



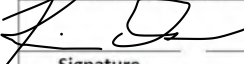
**HITACHI**

GE Hitachi Nuclear Energy

## Ontario Power Generation Inc. Darlington New Nuclear Project:

# BWRX-300 Preliminary Safety Analysis Report



<b>ONTARIOPOWER</b> GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
	15MAR2023
Signature	Date
Name: Karim Osman	
Dept: DNNP Design Engineering	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	
Notes:	
OPG Document Number: NK054-SR-01210-00001 R001	
OPG Security Classification: OPG Proprietary	

Revision 1  
March 7, 2023

Copyright 2023, GE-Hitachi Nuclear Energy Americas, LLC

All Rights Reserved



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33950

Revision 2

October 7, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 1  
Introduction and General Considerations**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	Section 1.5 Section 1.6.2	Incorporate corrections per customer acceptance review.
2	Section 1.5 Section 1.11	Incorporate corrections per customer acceptance review.

### ACRONYM LIST

Acronym	Explanation
ACI	American Concrete Institute
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CB	Control Building
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CSA	CSA Group
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defense Line
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EME	Emergency Mitigating Equipment
EOC	Emergency Operations Centre
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FSF	Fundamental Safety Function

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Acronym	Explanation
GEH	GE Hitachi Nuclear Energy
GSS	Guaranteed Shutdown State
HCU	Hydraulic Control Unit
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	ICS Pool Cooling and CleanUp System
ICRP	International Commission on Radiological Protection
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ISFSI	Independent Spent Fuel Storage Installation
ISO	International Organization for Standardization
LCH	Licence Conditions Handbook
LOCA	Loss-of-Coolant Accident
LTC	Licence to Construct
MCR	Main Control Room
NBS	Nuclear Boiler System
NEI	Nuclear Energy Institute
NERC	North American Electric Reliability Corporation
NFPA	National Fire Protection Association
NPP	Nuclear Power Plant
NRCC	National Research Council of Canada
NSCA	Nuclear Safety and Control Act
OER	Operating Experience Review
OPG	Ontario Power Generation
PCCS	Passive Containment Cooling System
PIE	Postulated Initiating Event
PLSA	Plant Services Area
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
RB	Reactor Building

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SAMG	Severe Accident Management Guideline
SCA	Safety and Control Area
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SFPE	Society of Fire Protection Engineers
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components
TB	Turbine Building
USNRC	U.S. Nuclear Regulatory Commission

## TABLE OF CONTENTS

<b>1.0</b>	<b>INTRODUCTION AND GENERAL CONSIDERATIONS .....</b>	<b>1-1</b>
1.1	Introduction .....	1-1
1.1.1	Format of the Safety Analysis Report .....	1-1
1.1.2	DNNP Project Delivery.....	1-5
1.1.3	PSAR Verification Scope .....	1-5
1.1.4	PSAR Limitation of Use .....	1-6
1.2	Project Implementation .....	1-7
1.3	Identification of Interested Parties Regarding Design, Construction and Operation .....	1-8
1.4	Information on the Plant Layout and Other Aspects .....	1-9
1.4.1	Site Layout and Infrastructure .....	1-9
1.4.2	DNNP Switchyard .....	1-11
1.4.3	Normal Heat Sink .....	1-11
1.4.4	Security Building .....	1-12
1.5	General BWRX-300 Power Block Description .....	1-13
1.5.1	Basic Technical Characteristics .....	1-13
1.5.2	Primary Drawings.....	1-16
1.6	Comparison with Other Plant Designs .....	1-19
1.6.1	Enhancements in Safety System Design .....	1-19
1.6.2	Industry Incident Reviews .....	1-20
1.7	Drawings and Other More Detailed Information.....	1-21
1.7.1	Reactor Pressure Vessel and Internals.....	1-21
1.7.2	Reactor Pressure Vessel Isolation Valves .....	1-21
1.7.3	Control Rod Drive System .....	1-22
1.7.4	Isolation Condenser System .....	1-22
1.7.5	Instrumentation and Control.....	1-22
1.7.6	Containment.....	1-22
1.7.7	Passive Containment Cooling System.....	1-23
1.7.8	Boron Injection System .....	1-23
1.7.9	Reactor Water Cleanup System .....	1-23
1.7.10	Shutdown Cooling System.....	1-23
1.7.11	ICS Pool Cooling and Cleanup System .....	1-23
1.7.12	Fuel Pool Cooling and Clean System .....	1-23
1.7.13	Containment Inerting System.....	1-23

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

1.8	Modes of Normal Operation of the Plant.....	1-27
1.9	Principles of Safety Management .....	1-28
1.9.1	References.....	1-28
1.10	Additional Supporting or Complementary Documents to the Safety Analysis Report .....	1-29
1.11	Conformance with Applicable Regulations, Codes and Standards.....	1-33
1.11.1	References.....	1-37
1.12	Appendix A – Darlington Nuclear Site and DNNP General Arrangement Drawings .....	1-38
1.13	Appendix B – Tables of Design Basis REGDOCs, Codes, and Standards.....	1-44



**LIST OF TABLES**

Table 1.5-1: Principal Characteristics of Interest for the DNNP BWRX-300 .....	1-14
Table 1.5-2: Approximate Dimensions of Power Block Buildings .....	1-16
Table 1.6-1: Comparison of BWRX-300 to Other NPP Types .....	1-20
Table 1.10-1: Topical Reports Incorporated by Reference .....	1-30
Table B1.11-1: List of Plant Level Design Basis Regulatory Documents.....	1-45
Table B1.11-2: List of Plant Level Design Basis Codes and Standards .....	1-46
Table B1.11-3: List of Plant Level Design Basis US Regulatory Documents.....	1-50

## LIST OF FIGURES

Figure 1.7-1: BWRX-300 Major Systems .....	1-24
Figure 1.7-2: BWRX-300 RPV and Internals .....	1-25
Figure 1.7-3: Isolation Condenser System.....	1-26
Figure 1.11-1: CNSC Regulatory Framework .....	1-34
Figure 1.11-2: GEH Requirements Hierarchy .....	1-36
Figure A1.1-1: Darlington Nuclear Site Regional Location .....	1-39
Figure A1.1-2: Darlington Nuclear Site (DNNP Proximity to DNGS) .....	1-40
Figure A1.4-1: DNNP BWRX-300 Facility Site Layout.....	1-41
Figure A1.4-2: DNNP Switchyard Site Plan .....	1-42
Figure A1.5-1: BWRX-300 Power Block Plan View at Elevation 0 .....	1-43

## **1.0 INTRODUCTION AND GENERAL CONSIDERATIONS**

### **1.1 Introduction**

The Darlington New Nuclear Project (DNNP) will provide a new Class I nuclear facility, as defined by the Nuclear Safety and Control Act (NSCA), that is a critical new source of clean electricity for the Province of Ontario's future energy needs and achieving Canada's commitment to be Net-Zero by 2050. The DNNP will be implemented at the existing Darlington Nuclear site that is owned and operated by Ontario Power Generation (OPG).

The Darlington Nuclear site (see Appendix A Figure A1.1-1), is located in the township of Darlington, on the north shore of Lake Ontario at Raby Head, approximately 70 km east of Toronto. The site is approximately 5 km southwest of the community of Bowmanville and 10 km southeast of the City of Oshawa. Immediately to the east of the site is St. Marys Cement limestone quarry and processing plant. The site is traversed by an east-west operating Canadian National (CN) railway and a 8.5m high berm that provides the site protection in the event of a railway accident. The site is also traversed by the Lake Ontario Waterfront Trail, which is a multi-use recreation trail extending from Niagara-on-the-Lake to the Quebec border along the north shores of Lake Ontario and the St. Lawrence River.

Currently, the Darlington Nuclear site (see Appendix A Figure A1.1-2) is home to the 3512 megawatt-electric Darlington Nuclear Generating Station (DNGS), comprised of four operating CANada Deuterium Uranium (CANDU) pressurized heavy water generating reactors, the Tritium Removal Facility (TRF) that serves all of Ontario's CANDU nuclear reactors, and the Nuclear Sustainability Services-Darlington that stores spent nuclear fuel from the DNGS. The DNNP site is in the eastern one-third of the site bounded by the site property limits to the east and north, by Lake Ontario to the south, and by Holt Road to the west.

The DNNP site is also owned by OPG. OPG is the holder of a Nuclear Power Reactor Site Preparation Licence 18.00/2031. This licence permits OPG to perform activities to prepare the DNNP site for the future placement of a nuclear facility. In December 2021, OPG announced that the selected technology for this nuclear facility to be the grid-scale BWRX-300 Small Modular Reactor (SMR), designed by GE-Hitachi Nuclear Energy Americas, LLC (GEH). The BWRX-300 is approximately 300 megawatt-electric in size and, is capable of preventing between 0.3 and 2 megatonnes of carbon dioxide emissions per year, depending on the kind of alternative power generation technology it is displacing. OPG also announced that it will submit a Licence to Construct (LTC) application in accordance with the NSCA, Class I Nuclear Facilities Regulations (SOR/2000-204) and CNSC REGDOC-1.1.2 by the end of 2022. Once granted, the LTC will permit the construction of one BWRX-300.

In accordance with paragraph 5(f) of SOR/2000-204, this Preliminary Safety Analysis Report (PSAR) supports OPG's LTC application and demonstrates the adequacy of the design of BWRX-300. This PSAR has been prepared collaboratively between GEH and OPG and in accordance with the guidance of the International Atomic Energy Agency (IAEA) as documented in their Specific Safety Guide No. SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants.

#### **1.1.1 Format of the Safety Analysis Report**

It is recognized by the CNSC Regulatory Framework and IAEA guidance that Safety Analysis Reports (SARs) are developed in an iterative manner to support the appropriate licensing activities at the appropriate time. Since this release supports OPG's LTC application, this version of the SAR is a PSAR and contains sufficient design information commensurate with the stage of the design progression to assess and demonstrate that the plant can be safely constructed. The

PSAR will be updated to a Pre-Operational Safety Analysis Report with more detailed design information to support OPG's Licence to Operate application at the appropriate future time.

The following describes the format of the DNNP SAR and includes a brief description of the content of each chapter. Information presented in each chapter is commensurate with its importance to nuclear safety and PSAR purposes.

### **Chapter 1: Introduction and General Considerations**

Information in this chapter describes the DNNP and the PSAR, including their purposes and objectives. It describes DNNP facilities at a high level and the national and international guidance applied to the BWRX-300.

### **Chapter 2: Site Characteristics**

Information in this chapter describes the characteristics of the DNNP site on which the BWRX-300 facility is planned to be constructed. The information represents the baseline data which is used to ensure that site-related uncertainties are addressed and dispositioned in the final design and safety assessment of the BWRX-300 facility. This chapter describes the geological, seismological, hydrological, meteorological, and geotechnical features of the DNNP site and the surrounding region. It also describes the site-specific natural and human-induced external hazards including radiological conditions due to external sources and their dispersion characteristics. Furthermore, this chapter describes present and projected population distribution and land use relevant to the safe design and operation of the BWRX-300 facility over its expected 60-year operational life.

### **Chapter 3: Safety Objectives and Design Rules for Structures, Systems, and Components**

This chapter is specific to the BWRX-300 and introduces the safety objectives and the safety strategy framework to meet those objectives for its design and construction at DNNP. Additionally, this chapter describes the design rules for classification of Structures, Systems, and Components (SSC) important to safety, and the design principles and safety requirements established for the BWRX-300.

### **Chapter 4: Reactor**

This chapter is specific to the BWRX-300 and describes the design of its reactor and fuel assembly in more detail. It also provided the relevant information on the reactor that demonstrates its capability to fulfil relevant safety functions throughout the design life in all plant states.

### **Chapter 5: Reactor Coolant System and Associated Systems**

This chapter is specific to the BWRX-300 and describes Nuclear Boiler System (NBS) and interfacing systems that form the Reactor Coolant Pressure Boundary (RCPB). Further, the information in this chapter demonstrates that the functional and structural integrity aspects of the various NBS SSC are designed with robustness, quality, independence, redundancy, and diversity to maintain adequate reactor coolant inventory during Anticipatory Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs).

### **Chapter 6: Engineered Safety Features**

This chapter is specific to the BWRX-300 and describes its engineered safety features provided to mitigate the consequences of AOOs and DBAs. The engineered safety features are divided into three general groups: (1) Containment and Associated Systems; (2) Isolation Condenser System (ICS) functioning as the Emergency Core Cooling System; and (3) Control Room Habitability.

### **Chapter 7: Instrumentation and Control**

This chapter is specific to the BWRX-300 and describes its Instrumentation and Control (I&C) systems. The information in this chapter is organized to systematically present the I&C design bases in the necessary context to support an understanding of the individual I&C system designs and safety features.

### **Chapter 8: Electrical Power**

This chapter is specific to the BWRX-300 and describes its electrical system and requirements and how they interface with Darlington Nuclear site and Ontario's electrical transmission system.

### **Chapter 9A: Auxiliary Systems**

This chapter is specific to the BWRX-300 and describes its auxiliary systems (e.g., fuel handling, water, air, HVAC, fire protection, diesel generators, cranes, etc.) that support its safe and reliable operations.

### **Chapter 9B: Civil Engineering Works and Structures**

This chapter is specific to the BWRX-300 and describes how its general seismic design requirements are complied with in the design of the Reactor Building (RB). Information is also provided describing the general civil and structural design requirements for the Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), Plant Service Area (PLSA), Intake and Discharge Structures, Forebay and tunnels to and from the lake and Fire Pump Enclosure.

### **Chapter 10: Steam and Power Conversion Systems**

This chapter is specific to the BWRX-300 and describes design of the power conversion system, including the Main Turbine Equipment, which is comprised of the High Pressure and Low Pressure Turbines, Turbine Gland Seal System, Turbine Lubricating Oil System, Extraction Steam System, Electro-Hydraulic Controls System, and Turbine Auxiliary Steam Subsystem.

### **Chapter 11: Management of Radioactive Waste**

Information in this chapter describes the main sources of liquid, gaseous and solid radioactive waste including the radiological source term used in calculating liquid and airborne effluent. Also described are the radioactive waste processing systems (i.e., pretreatment, treatment, and conditioning systems) as well as temporary waste storage located on the site. The SSCs that monitor and sample the process and effluent streams to measure and control the discharge of radioactive materials generated in operational states and accident conditions are described.

The measures proposed for the safe management of radioactive and hazardous waste of all types that will be generated throughout the lifetime of the plant as well as how these measures meet the relevant safety requirements including the measures taken for the safe management and disposal of this waste are described in OPG's Waste Management Program.

### **Chapter 12: Radiation Protection**

Information in this chapter describes the administrative programs and procedures, in conjunction with facility design, that ensures occupational radiation exposure to personnel is kept As Low As Reasonably Achievable (ALARA). The systematic application of the ALARA philosophy during the design phase of the BWRX-300 that establishes the basic design criteria observed to reduce occupational exposure during plant operation and maintenance, decommissioning and post-accident ALARA are described.

### **Chapter 13: Conduct of Operations**

Information in this chapter describes how OPG fulfils its prime responsibility for safety in the operation of the BWRX-300. Specifically, this chapter describes important operational issues relevant to nuclear safety, approaches adopted by OPG to address these issues through its operational programs and provisions made by OPG that establish and maintain an adequate number of qualified staff. The preparation of OPG operating procedures for the BWRX-300 that ensure its safety is supported by GEH.

### **Chapter 14: Plant Construction and Commissioning**

Information in this chapter describes how OPG assures that the BWRX-300 will be suitable for service prior to entering the construction, commissioning, and operational stages. The commissioning program and organization intended to verify and validate the plant's performance against the design prior to the turnover of the facility to OPG for operation are described. Additionally, information is provided on how OPG qualified operating personnel at all levels are trained and directly involved in the commissioning process.

### **Chapter 15: Safety Analysis**

This chapter is specific to the BWRX-300 and describes the results from the BWRX-300 plant safety analyses that includes the Deterministic Safety Analysis, Severe Accident Analysis, Hazard Analysis and Probabilistic Safety Assessment (PSA). It describes how the safety analysis verifies, throughout the iterative design and analysis process, that the design of the BWRX-300 adequately performs the Fundamental Safety Functions (FSF) of controlling reactivity, fuel cooling, long-term heat removal, and containment of radioactive materials.

### **Chapter 16: Operational Limits and Conditions for Safe Operation**

Information in this chapter describes how the facility's safe operating envelope is evaluated and implemented through a set of operational limits and conditions that prescribe boundaries within which OPG must operate the BWRX-300 to assure compliance with the safety analysis inputs, assumptions, and results. The full set of operational limits and conditions are a key element of the licensing basis for a Licence to Operate.

### **Chapter 17: Management for Safety**

Information in this chapter describes how the overall management of all safety related activities is assured throughout the entire lifecycle of the facility. It describes the general, specific, quality management, performance improvement, and safety culture elements of the management systems of OPG and the organizations, such as GEH, that support the development, operation, and eventual retirement of DNNP facilities.

### **Chapter 18: Human Factors Engineering**

This chapter is specific to the BWRX-300 and describes the Human Factors Engineering program for the development lifecycle phases (i.e., design, construction, and commissioning) of the BWRX-300. This chapter demonstrates the effective integration of Human Factors Engineering requirements and analysis results into the design of the plant in an iterative process.

### **Chapter 19: Emergency Preparedness and Response**

Information in this chapter demonstrates that in a very unlikely nuclear or radiological emergency occurring at the DNNP facility, timely and effective actions are taken that protect workers, the public, and the environment coordinated with off-site government agencies and supported by a documented decision-making process.

## **Chapter 20: Environmental Aspects**

Information in this chapter describes the environmental aspects important for the development, operation, and retirement of the DNNP facilities. General aspects of the Environmental Impact Assessment, applicable principles, and regulations, OPG's Environmental Management System, and site characteristics are described. Features that minimize environmental impact of the facility throughout its entire lifecycle, including postulated accidents, are also described.

## **Chapter 21: Decommissioning and End of Life Aspects**

Information in this chapter demonstrates their commitment to the production of energy in a sustainable manner through the effective and efficient life-cycle management of the nuclear facilities. The planning for decommissioning and end-of-life management of DNNP facilities, including the BWRX-300 reactor, is an integral aspect of the facility life-cycle management process that is described in this chapter.

## **Safeguards Annex: Safeguards and Nuclear Material Accountancy**

Information is presented in the Safeguards Annex to demonstrate how OPG supports compliance with the IAEA safeguards agreement in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons and the Additional Protocol to the Safeguards Agreement. Under the NSCA, the CNSC has the mandate to achieve Canadian conformity with such international obligations. Specifically, this Safeguards Annex describes information related to the BWRX-300 reactor facility at the DNNP to demonstrate compliance with the international agreements, as well as compliance with the responsibilities included in the NSCA and the General Nuclear Safety and Control Regulations (SOR-2000-202).

## **Security Annex: Darlington BWRX-300 Security Assessment**

Detailed information about the protected area and vital areas, including their structures and/or barriers, are provided in a separate security annex since the content contains prescribed information as defined by Section 21 of the General Nuclear Safety and Control Regulations (SOR/2000-202).

### **1.1.2 DNNP Project Delivery**

OPG will implement DNNP through a project delivery model described in Section 1.2 having a target in service date as early as 2028. The DNNP is composed of the following five key projects that will assure its safe and efficient implementation:

1. Site Preparation Project
2. Power Block Project
3. Switchyard & Grid Connection Project
4. Intake / Discharge Water System Project
5. Digital Strategy Project

### **1.1.3 PSAR Verification Scope**

The PSAR is a licensing basis document that is prepared, validated, and approved in accordance with GEH's Quality Assurance Program, which assures:

1. Accuracy of information against verified engineering source documentation
2. Sufficient information to support the Licence to Construct (LTC) in accordance with CNSC REGDOC-1.1.2



NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

3. Alignment with the stage at which the design has progressed and its supporting safety analysis

**1.1.4 PSAR Limitation of Use**

The PSAR is a high-level summary document developed to support the LTC application activities for DNNP only. It is released for information and licensing purposes only and shall not be used for technical or construction purposes.

## **1.2 Project Implementation**

For the DNNP development lifecycle phase, OPG will utilize an Integrated Project Delivery contract model which maximizes integration and collaboration with other contract partners involved with this phase. The Integrated Project Delivery contract agreement describes the relationship and accountabilities of the contract partners including Owner (OPG), Developer (GEH), Constructor, and Architect Engineer.

Key principles of the contract model include collaborative behaviours, common information systems, best athlete approach to staffing positions, risk and reward sharing, transparency between partners, and maximizing efficiencies.

Roles and responsibilities for all contract partners are defined and accepted through contractual agreements for the project. The partner roles are further described in Chapter 17, Subsection 17.2.1.

### **1.3 Identification of Interested Parties Regarding Design, Construction and Operation**

The Developer, GEH, is the Design Authority for scoped design activities in accordance with the project execution model until turnover.

The Constructor is responsible for procurement, construction, and support of commissioning activities.

An Architect Engineering firm performs design and engineering activities required for the project.

The owner and the licence applicant, OPG, is ultimately responsible for DNNP operation and retains the overall accountability for ensuring the project lifecycle is executed with quality and safety.

The contractual agreement between the stakeholders details specific responsibilities and interfaces are further detailed in Chapter 17.

## **1.4 Information on the Plant Layout and Other Aspects**

The DNNP site layout, infrastructure, intake and discharge water, Switchyard, BWRX-300 Power Block, and their respective interfaces are shown in Appendix A (see Figures A1.4-1 and A1.4-2).

Descriptions of the site layout, infrastructure, intake and discharge water, and Switchyard are provided in this Section.

Descriptions of the BWRX-300 Power Block are provided in Section 1.5.

### **1.4.1 Site Layout and Infrastructure**

The DNNP site will contain infrastructure, including additional buildings, to support operations inside the Power Block.

Currently anticipated services include:

1. A demineralized water supply pipeline extending from the Darlington Demineralized Water Plant eastwards approximately 400m towards the DNGS/DNNP property line along the Third Line Road corridor. The demineralized water is used for the Power Block operations.
2. A potable water pipeline extension tying into the existing municipal water supply just south of the CN railway and west of Holt Road bridge on the west side of the road. This pipe carries potable water for use inside the power block as well as various outbuildings around the DNNP property including the administration building, warehouse, temporary construction buildings, and potentially other buildings to be determined.
3. Sanitary sewer connections to the existing Darlington East Sewage Lift Station are planned. These carry sewage from inside the power block as well as the administration building, warehouse, and potentially other buildings not yet determined, to the lift station. From here the effluent is pumped north and west towards the Courtice Water Pollution Control Plant for treatment and eventual discharge to Lake Ontario.
4. Fibre-optic cables for a business Local Area Network and copper telephone/public address cables to create a communications link between DNGS and DNNP. These run from the DNGS Engineering Support Services Building in an underground duct bank eastwards approximately 400m towards the DNGS/DNNP property line mostly along the Second Line Road corridor.
5. Additional fibre-optic cables for a security Local Area Network are brought from the Darlington Main Security Building approximately 600m east towards the DNGS/DNNP property line in an underground duct bank mostly along the Second Line Road corridor.
6. Up to 10MW of construction power are brought from the existing 54M15 feeder through Darlington DS5 substation at 13.8kV, located near the intersection of Park Road and Second Line, approximately 1km east to a new switchgear to be located near the northeast corner of the Nuclear Sustainability Services-Darlington. This switchgear is planned to feed construction loads as well as the new administration building and warehouse. A second feed will be taken from the same 54M15 through the existing Darlington DS1-F1 substation at 8.32kV and will supply construction loads including the construction trailers.

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Planned structures/features include:

1. An administration building with office spaces and a simulator training space. The simulator space will support the SMR full scope simulator and desktop simulator plus limited maintenance training space.
2. A warehouse is necessary to provide long term storage space for SMR components and equipment. It has some maintenance space and a calibration shop suitable for the service of non-contaminated equipment.
3. There is a parking lot near the administration building. There is an existing parking lot south of the Canadian National Railway near the border between DNNP and St. Marys Cement that will also be utilized.
4. A Steel Bricks production facility is planned to be constructed on the northwest quadrant of the intersection of Maple Grove Road and Second Line. This facility produces the Steel Bricks components for the construction of the reactor building.
5. A concrete batching plant will be provided suitably located if it is determined that onsite concrete batching is required.
6. Holt Road will be improved in two phases:
  - a. Phase 1 - The Holt Road extension is a new stretch of road to be built from the intersection of Second Line and Old Holt Road at the northwest corner of DNNP property. This will extend south along the DNGS/DNNP property line between the Nuclear Sustainability Services-Darlington and the SMR facility until it reaches Lake Ontario. At this point it turns west and continues until it connects with the existing Lakeshore Road. The portion of Holt Road along Lake Ontario will be reinforced, and form part of the heavy haul route used to transport heavy components from the DNGS wharf to DNNP.
  - b. Phase 2 - The Holt Road expansion will add an additional northbound lane from Second Line north towards Highway 401. This additional lane will end south of Energy Drive and will be used by soil transport trucks to place soil onto the northern parts of DNNP property forming the spoils piles. There will also be a new southbound left turn lane to be created just south of the Holt Road bridge to aid traffic turning onto DNNP property.
7. The existing Old Holt Road that stretches diagonally across DNNP property will be kept intact up to the point it joins the ring road around the Power Block facility.
8. The heavy haul road along Lakeshore Road will extend east onto DNNP property to support the construction of the Power Block. It is planned to extend as far east as the Power Block facility and then extend only as far north as necessary to support the Power Block facility construction.
9. Maple Grove Road is planned to be improved and extended south and then west to join the heavy haul road at the south part of the DNNP property. The improvements will likely include a new bridge to cross over the Canadian National Railway.
10. A soil conditioning pile is created from excavated earth during the site preparation phase and located at the southeast quadrant of the Maple Grove Road and Second Line intersection. This soil will be reconditioned and placed back into the SMR facility foundation.

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

11. A soil spoils pile is located in the northern part of DNNP property south of Energy Drive and west of Maple Grove Road. Excavated earth from the site preparation phase will be placed here.
12. Storm water management features are part of the overall site layout. One known feature is the relocation of the existing Bowmanville SS drainage ditch that currently runs from Bowmanville SS through DNNP property, southeast along Old Holt Road and draining into Lake Ontario. This will be relocated west to run along the eastern edge of the new Holt Road Extension.

#### **1.4.2 DNNP Switchyard**

The local DNNP switchyard (see Appendix A, Figure A1.4-2) is located North of the SMR Facility, East of the Extended Holt Rd and South of the CN Rail tracks. The local switchyard consolidates power produced by the Power Block facility. The Power Block facility has two 230kV lines connected to the local DNNP switchyard. One line connects the Facility Generator Step Up Transformer, and one line connects to the Reserve Auxiliary Transformer. The local switchyard has two redundant 230kV connections with the transmitter. The transmitter is working to connect these lines to Clarington TS, 22km North of the DNNP site.

The operating organization is responsible for the ownership and operation of the local DNNP switchyard, containing the high voltage circuit breakers and disconnect switches, in addition to equipment within the Power Block facility. Hydro One, the transmitter for the electrical grid, is responsible for the ownership and operation of the two redundant 230kV lines connecting the local DNNP switchyard with Clarington Transformer Station.

#### **1.4.3 Normal Heat Sink**

The normal heat sink removes excess heat to a large water body. For the DNNP, water withdrawn from Lake Ontario flows through the plant surface condensers to remove the excess energy of the turbine exhaust steam. The amount of heat removed during this process depends on the flow rate and the temperature rise of the water passing through the condensers. The plant heat is rejected to Lake Ontario.

Cooling water from Lake Ontario is delivered to an intake structure for the nuclear facility through an intake tunnel. The intake structure sends the cooling water to the Pumphouse/Forebay that contains circulating water pumps which deliver the cooling water to plant surface condensers before returning the heated water back to the lake through the discharge tunnel.

The Normal Heat Sink includes, but is not limited to the following:

1. Intake Tunnel, located deep in Lake Ontario to meet regulatory commitments (D-C-1) to decrease potential impacts to fish habitat and is sized to provide the required flow of cooling water to the plant. It is also constructed to minimize the intake velocity to prevent impingement and entrainment of fish and effect on local currents.
2. Discharge Tunnel and diffusers are constructed deep in Lake Ontario to meet regulatory requirements by limiting the temperature increase to minimize thermal and flow effects of the plant cooling water discharge to ensure surface water temperature does not exceed 2 degrees C above ambient surface temperature and minimize impact to aquatic habitat.
3. Pumphouse/Forebay is composed of the forebay, pump bays and superstructures to house the Circulating Water System pumps and related equipment.

#### **1.4.4 Security Building**

A security building, known as the Protected Area Access Building, is provided on the protected area boundary to allow for ingress and egress to and from the protected area. Additionally, a sally port is provided adjacent to the security building to allow for vehicular traffic to enter the protected area. Detailed information about the protected area and vital areas, including their structures and/or barriers, are provided in a separate security annex since the content contains prescribed information as defined by Section 21 of the General Nuclear Safety and Control Regulations (SOR/2000-202).



## **1.5 General BWRX-300 Power Block Description**

The BWRX-300 is a Boiling Water Reactor (BWR) that employs natural circulation and passive emergency cooling features and is rated at approximately 300 megawatts-electric.

The passive design features of the BWRX-300 provide decay heat removal capability using only installed systems with no reliance on operator actions or external resources for at least 72-hours. For the BWRX-300, a safe stable condition ("stable shutdown") is defined as safe shutdown with average reactor coolant temperature  $\leq 215.6^{\circ}\text{C}$  ( $420^{\circ}\text{F}$ ). Following 72-hours post-accident, on-site or off-site resources are used to power non-safety equipment for proceeding to cold shutdown conditions, as needed.

The BWRX-300 design applies a defence-in-depth process for safety assessment and safety analysis to ensure that radiological acceptance criteria are met. The leveraging of passive design features greatly simplifies the design and results in a significant reduction in total number active SSCs compared to conventional Nuclear Power Plants (NPPs).

The overall safety objectives and the safety strategy employed in the development of the BWRX-300 design are described in detail in Chapter 3.

### **1.5.1 Basic Technical Characteristics**

The principal technical characteristics of the BWRX-300 are provided in Table 1.5-1.

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 1.5-1: Principal Characteristics of Interest for the DNNP BWRX-300**

Parameter Description	Value	Comments
Type of plant	Boiling Water Reactor	
Core coolant	Light Water	
Neutron moderator	Light Water	
Nuclear Steam Supply System layout	Direct-Cycle	
Primary circulation	Natural	
Thermodynamic cycle	Rankine	
Type of containment structure	Dry	
Reactor thermal power level	870 MWth	
Normal Heat Sink	Once Through Cooling System using water from Lake Ontario	
Ultimate Heat Sink	ICS pools	Pools are vented to atmosphere
Plant gross electrical power output	~ 300 MWe	
Plant Footprint	~ 9,800 m <sup>2</sup>	Rectangular building envelope
Site Footprint	~ 30,000 m <sup>2</sup>	Fenced area
Design life	60 years	
Exclusion Zone	350 m (radius)	Measured from exterior of the Reactor Building
Seismic Design (DBE)	0.310 g (horizontal and vertical)	Bounding rock peak ground acceleration
	0.532 g (horizontal)	Bounding surface peak ground acceleration
	0.516 g (vertical)	
Reactor Design Pressure	10.3 MPa	
Fuel	UO <sub>2</sub> pellets	
Fuel enrichment	<5% U-235	
RPV diameter (ID)	~ 4 m	
RPV height (Inside)	~ 26 m	
Control rod drive type	Fine Motion Control Rod Drive (FMCRD)	
Containment Vessel type	Steel-plate Composite	
Fuel pool capacity	Up to 8 years of full-power operation	Fuel pool accommodates up to 8 years of spent fuel plus one core load of new fuel and one full core off-load
Refueling cycle	12 - 24 months	

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Parameter Description	Value	Comments
Emergency Power Supply	Safety Class 1 DC batteries	Capable of sustaining required loads for 72 hours

### 1.5.2 Primary Drawings

An overview of the primary buildings/structures in the Power Block of the BWRX-300 is shown in Figure A1.5-1 of Appendix A and is discussed below. The Reactor Building, Turbine Building, Radwaste Building, Control Building, and Plant Services Area (PLSA) are specifically referenced in the descriptions below.

The plant grade elevation of the power block is approximately 88 meters Canadian Geodetic Vertical Datum of 1928. The overall Power Block length is approximately 143 meters and width is approximately 69 meters. The approximate dimensions of the power block buildings are provided in Table 1.5-2.

**Table 1.5-2: Approximate Dimensions of Power Block Buildings**

Building	Length (m)	Width (m)	Highest Roof Elevation (m)
Reactor Building <sup>(1)</sup>	36 (Diameter)	36 (Diameter)	30 (Exterior Dome Top)
Turbine Building	70	69 <sup>(3)</sup>	30 <sup>(4)</sup>
Radwaste Building <sup>(2)</sup>	38	25	24 <sup>(5)</sup>
Control Building	35	69 <sup>(3)</sup>	10 <sup>(5)</sup>
Reactor Auxiliary Bay <sup>(6) +</sup>	38	18 <sup>(7)</sup>	10 m at the highest roof

- (1) The bottom elevation of the Reactor Building foundation is approximately 36 m below grade.
- (2) The Radwaste Building wraps around the Reactor Building. Width of Radwaste Building is given as the shortest dimension of the building measured in the east-west direction.
- (3) The Turbine Building and Control Building width include portions of the Plant Services Area.
- (4) The height of the Turbine Building does not include the stacks or stairwells.
- (5) The height of the Radwaste Building and Control Building does not include chillers, ductwork, or other items on the roof.
- (6) The Reactor Auxiliary Bay is a portion of the Plant Services Area, to the east of the Reactor Building, that is supported on an independent foundation with respect to the surrounding Reactor Building, Control Building, and Turbine Building.
- (7) The Reactor Auxiliary Bay wraps around the Reactor Building. The width of Reactor Auxiliary Bay is given as the largest dimension of the building measured in the east-west direction.
- (8) For consistency, the length and width values in the table are all in the same direction. For DNNP-1, length is north-south, and width is east-west.

Refer to Chapter 3, Subsection 3.3.1, and Chapter 9B, Section 9B.2 and 9B.3 of the PSAR for additional information on seismic design of structures.

#### **1.5.2.1 Reactor Building**

The RB is a Safety Category 1 and Seismic Category A structure. It is a cylindrical shaped structure embedded in a vertical shaft to a depth of approximately 36 m below grade. The Reactor Pressure Vessel (RPV), Steel-plate Composite Containment Vessel (SCCV) and other important systems and components are located in the deeply embedded RB vertical right-cylinder shaft to mitigate effects of external events, including aircraft impact, adverse weather, fires, and earthquakes. The Secondary Control Room (SCR) is located in the RB. The below-grade portion also contains reactor support and safety class systems and the Safety Class 1 power supply and equipment. The reactor cavity pool is above the containment dome. Also, within the RB, three separate Isolation Condenser System (ICS) pools sit next to the reactor cavity pool above the SCCV, with one isolation condenser located in each pool. The Fuel Pool is also located in the RB.

#### **1.5.2.2 Turbine Building**

The TB houses the steam turbine generator, standby diesel generators, main condenser, condensate and feedwater systems, turbine-generator support systems, and parts of the Offgas System (excluding the offgas charcoal adsorbers).

While considered a separate functional area from the Turbine Building, the northern portion of the PLSA is structurally integrated with the Turbine Building. See Section 1.5.2.5 for a description of the PLSA.

The TB is a Safety Class 2 structure and is categorized as Non-Seismic. Additionally, it is evaluated for seismic interaction to ensure that it will not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a Design Basis Earthquake (DBE) or extreme Tornado wind conditions.

#### **1.5.2.3 Control Building**

The CB houses the MCR, Emergency Operations Centre (EOC), electrical, control, and instrumentation equipment. The CB is a Safety Class 2 structure and is categorized as Non-Seismic. Additionally, it is evaluated for seismic interaction to ensure that it does not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a DBE or extreme Tornado wind conditions. The CB serves as main entrance and exit for the Power Block unit during normal operations.

While considered a separate functional area from the Control Building, the southern portion of the Plant Services Area (PLSA) is structurally integrated with the Control Building. See Section 1.5.2.5 for a description of the PLSA.

#### **1.5.2.4 Radwaste Building**

The RWB houses rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes as well as the Offgas System charcoal adsorbers that are used for processing radioactive gas. Some of these systems contain highly radioactive materials. The RWB is classified as a Safety Class 3 building and categorized as RW-IIa in accordance with Regulatory Guide (RG) 1.143, Rev. 2, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water-Cooled Nuclear Power Plants." Additionally, it is also evaluated for seismic interaction to ensure that it will not compromise the structural integrity or safety functions of the adjacent Seismic Category A RB following a DBE or extreme Tornado wind conditions.

#### **1.5.2.5 Plant Services Area and Reactor Auxiliary Bay**

The PLSA houses a decontamination area, a contaminated part/tool storage room, a I&C calibration room, a truck space for cask removal, a hot machine shop, laydown areas for new fuel and the Fine Motion Control Rods Drives (FMCRD), and a miscellaneous storage area.

While the PLSA is a separate functional area from the CB and TB, the northern portion of the PLSA shares a foundation and is structurally integrated with the TB and the southern portion of the PLSA shares a foundation and is structurally integrated with the CB (see Figure A1.5-1).

A portion of the PLSA, the Reactor Auxiliary Bay, is constructed on a separate foundation with respect to the portions of the PLSA that are adjacent to the CB and TB. The functions performed in the Reactor Auxiliary Bay include new fuel and spent fuel cask transit, equipment ingress and egress to the RB, and personnel access to the RB. The Reactor Auxiliary Bay is a Safety Class 2 structure and is categorized as Non-Seismic. Additionally, it is evaluated for seismic interaction to ensure that it does not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a DBE or extreme Tornado wind conditions.

## **1.6 Comparison with Other Plant Designs**

The BWRX-300 is based on the U.S. Nuclear Regulatory Commission (NRC) licensed, 1520 MWe Economic Simplified Boiling Water Reactor (ESBWR). The ESBWR is an evolution of the 600 MWe Simplified Boiling Water Reactor (SBWR) that has a significant testing and qualification program directly applicable to the BWRX-300.

The BWRX-300 is the tenth generation of the Boiling Water Reactor (BWR) that incorporates the lessons learned in design, construction, operations, and maintenance from over 100 previous BWRs that have been built, operated, and in some cases, decommissioned.

The BWRX-300 is specifically designed to enhance safety through simplification and reducing its dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems. These safety enhancements, in combination with its reduction in scale and complexity, enable the reductions in operating staff, maintenance, and security requirements as well as being easier to decommission.

The BWRX-300 provides clean and flexible baseload electricity at a lifecycle cost that is much lower than the previous generation of NPPs operating today and competitive with other forms electricity generation such as natural gas combined-cycle plants and renewables.

### **1.6.1 Enhancements in Safety System Design**

Though mostly traditional in BWR design, the BWRX-300 includes several design features that simplify the design and enhance safety, such as:

1. **Reactor Isolation Valve location:** The BWRX-300 RPV is equipped with Reactor Isolation Valves which rapidly isolate a ruptured pipe to help mitigate the effects of a LOCA. All large fluid pipe systems are equipped with double isolation valves which are integral to the RPV. The valves are located as close as possible to the RPV.
2. **No Safety Relief Valves:** Safety relief valves have been eliminated from the BWRX-300 design. The large capacity Isolation Condenser System (ICS) provides overpressure protection in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Class 1 equipment. Historically on BWRs, the safety relief valve inadvertent actuation has been the most likely cause of a LOCA and have, therefore, been eliminated from the BWRX-300 design.
3. **Dry containment:** The BWRX-300 has a dry containment that is cooled through natural circulation during DBAs. This has been proven to effectively contain the releases of steam, water, and fission products after a LOCA.
4. **No external reactor recirculation loops:** Elimination of external reactor recirculation pumps and associated piping and a reimagined reactor pressure vessel provides a relatively large inherent reactor coolant volume and nozzle elevations significantly above the core. These features with a reliable passive emergency core cooling system provided by the isolation condensers eliminates the need for active emergency core cooling injection systems while ensuring larger safety margins than predecessor BWRs.
5. **No need for emergency diesel generators:** Elimination of active emergency core cooling systems eliminates the need for onsite emergency power systems. Standby diesel generators are provided for asset protection only.

Table 1.6-1 demonstrates how the BWRX-300 design has evolved to maximize passive safety features to achieve FSF in comparison to the design of previous BWR and other types of NPPs.



**Table 1.6-1: Comparison of BWRX-300 to Other NPP Types**

<b>Fundamental Safety Function</b>	<b>BWRX-300</b>	<b>BWRs</b>	<b>PWRs</b>	<b>CANDU</b>
Control Reactivity	Two Independent Means of Shut Down	Two Independent Means of Shut Down	Two Independent Means of Shut Down	Two Independent Means of Shut Down
Fuel Cooling	Passive Natural Circulation	Active Forced Circulation	Active Forced Circulation	Active Forced Circulation
Contain Radioactivity	Dry  Passive Cooling	Wet  Active Cooling	Dry  Active Cooling	Dry Reactor Building Wet Vacuum Building Active Reactor Building Cooling

## 1.6.2 Industry Incident Reviews

Station Blackout events have historically been the most demanding for BWRs to cope with and have usually been the dominant sequence for Severe Accident scenarios. The BWRX-300 is an advanced passive reactor design that does not require active safety systems. The BWRX-300 design carried forward the passive ICS and containment cooling concepts from the ESBWR. DC power sources are assumed to be available. The systems that support FSF and plant monitoring are designed to operate for 72-hours, without AC power, and without an intake structure that normally provides cooling water. The ICS pools and spent fuel pool have enough inventory to provide adequate decay heat removal and fuel cooling for seven days, after which alternate water makeup sources (e.g., flexible mitigation/EME) are used to refill the pools. The Passive Containment Cooling System (PCCS) is designed to passively limit containment pressure and temperature by transferring heat to the equipment pool. The demonstration of plant safety functions during a beyond design basis external event such as an earthquake that creates these conditions is typically part of the diverse and flexible coping strategies that form the basis for compliance of regulatory requirements related to the Fukushima tsunami event.

In April 2012, the Institute of Nuclear Power Operations conducted an independent review of the Fukushima nuclear accident with the purpose of identifying operational and organizational lessons learned from the accident. The results of this review are well documented.

The Fukushima accident was a Beyond Design Basis event. Design extension conditions are a selected subset of Beyond Design Basis accident conditions.

The BWRX-300 is designed for Design Extension Conditions, and these are described in detail in the BWRX-300 Safety Strategy.

## **1.7 Drawings and Other More Detailed Information**

A simplified representation of the major BWRX-300 systems and the flow of the reactor coolant is provided in Figure 1.7-1. A summary description of the major nuclear steam supply systems and components is provided below. Each of these systems are described in detail in applicable chapters of the PSAR.

### **1.7.1 Reactor Pressure Vessel and Internals**

The RPV is a vertical, cylindrical pressure vessel fabricated with forged rings and rolled plate welded together, with a removable top head, head flange, seals, and bolting. The vessel also includes penetrations, nozzles, and the shroud support. The RPV has a minimum inside diameter of approximately 4 m, a wall thickness of approximately 14 cm with cladding, and a height of approximately 26 m. The bottom of the active fuel region is approximately 5.2 m from the bottom of the vessel and the active core is 3.8 m high. The vertical orientated and tall vessel permits the development natural circulation driving forces to produce sufficient core coolant flow.

A diagram of the BWRX-300 RPV assembly is shown in Figure 1.7-2. The RPV, together with its internals, provides guidance and support for the Fine Motion Control Rod Drives (FMCRDs).

The major reactor internal components include:

- Core (fuel, channels, control rods and instrumentation)
- Core support and alignment structures (shroud, shroud support, top guide, core plate control rod guide tube, CRD housings, and orificed fuel support)
- Chimney
- Chimney head and steam separator assembly
- Steam dryer assembly
- Feedwater spargers
- In-core guide tubes

The fuel assemblies (including fuel rods and channels), control rods, chimney head, steam separators, steam dryer, and in-core instrumentation assemblies are removable when the reactor vessel is opened for refueling or maintenance. The RPV shroud support is designed to support the shroud, as well as the components connected to the shroud, including the steam separator, chimney, core plate, and top guide. The fuel bundles are supported by the orifice fuel support, the control rod guide tube, and the CRD housing.

### **1.7.2 Reactor Pressure Vessel Isolation Valves**

The BWRX-300 reactor incorporates isolation valves attached directly to the RPV. The function of the isolation valves is to close, limiting the loss of coolant from large and medium pipe breaks. The RPV isolation concept consists of two Reactor Isolation Valves in series. Each of the Reactor Isolation Valves is independently able to isolate the line.

### **1.7.3 Control Rod Drive System**

The CRD system includes three major elements: FMCRD mechanisms; Hydraulic Control Unit (HCU) assemblies; and the Control Rod Drive Hydraulic subsystem. The FMCRDs are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals. The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. In addition to hydraulic-powered scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic-powered scram.

### **1.7.4 Isolation Condenser System**

The ICS removes decay heat after any reactor isolation and shutdown event during power operations. The ICS decay heat removal limits increases in steam pressure and maintains the RPV pressure and water inventory at an acceptable level. The ICS consists of three independent loops that each contain a Heat Exchanger (HX) with capacity of approximately 33 MW, or approximately 3.7% of rated thermal power. Thermal energy removal condenses steam on the tube side and transfers heat by heating/evaporating water in the Isolation Condenser (IC) pools which are vented to atmosphere. The arrangement of the ICS HX is shown in Figure 1.7-3.

### **1.7.5 Instrumentation and Control**

The BWRX-300 Distributed Control and Information System (DCIS) is an integrated control and monitoring system for the power plant. The DCIS is arranged in three safety classified DCIS segments and a non-safety class segment with appropriate levels of hardware and software quality corresponding to the system functions they control and their DL location, as described in Chapter 3, Section 3.2. The DCIS provides control, monitoring, alarming and recording functions. The various bus segments of the integrated DCIS are designed to operate autonomously.

Control of reactivity in various postulated events is achieved by the instrumentation and control systems. Channels, trip logic, trip actuators, manual controls, and scram logic circuitry initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The hydraulic scram is actuated on signals derived from safety analyses and includes signals such as high core neutron power, RPV pressure, low RPV level, high containment temperature, high steam line flow, etc.

### **1.7.6 Containment**

The BWRX-300 Primary Containment Vessel encloses the RPV and some of its related systems and components. The Primary Containment Vessel is a leak-tight nitrogen inerted gas space surrounding the RPV and the Reactor Coolant Pressure Boundary (RCPB). It provides a leak-tight barrier to prevent the release of radioactive fission products, steam, and water in the unlikely event of a Loss of Coolant Accident (LOCA). The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The containment shape is a vertical cylinder approximately 18 meters outside diameter and 38 meters high. It is integral to and surrounded by the Reactor Building (RB).

### **1.7.7 Passive Containment Cooling System**

The Passive Containment Cooling System (PCCS) is a passive containment heat removal system that maintains the containment within its pressure limits for design basis accidents such as a LOCA. It consists of several low-pressure natural circulation heat exchangers that transfer heat from the containment to the reactor cavity pool which is located above the containment upper head and is filled with water during normal operation. The reactor cavity pool is vented to the atmosphere. PCCS operation requires no sensing, control, logic, or power actuated devices for operation.

### **1.7.8 Boron Injection System**

The Boron Injection System (BIS) is a complementary design feature that provides an additional means to place the plant in a cold shutdown mode. The BIS provides an additional means of negative reactivity insertion to bring the reactor subcritical during events when the control rod insertion (hydraulic and motor) is not successful.

### **1.7.9 Reactor Water Cleanup System**

The Reactor Water Cleanup System provides the design functions of a cleanup flow path from the RPV to filter/demineralizer skids during most reactor operating modes. The cleanup or filtration function and ion removal function is performed by the condensate system.

### **1.7.10 Shutdown Cooling System**

The Shutdown Cooling (SDC) System is designed to support RPV Startup and Shutdown/Cooldown Operations. The SDC consists of two independent trains with a motor driven pump, a heat exchanger, required valves, piping, controls, and power inputs.

### **1.7.11 ICS Pool Cooling and Cleanup System**

The Isolation Condenser Pool Cooling and Cleanup System (ICC) is designed to maintain the water in the ICS pools cool and clean.

The primary function of the ICC is to remove heat from the Isolation Condenser System (ICS) pools such that the bulk temperature of water in the pools is maintained below prescribed limits, and thereby ensure the readiness of the ICS to perform its safety function. Secondary functions of the ICC include maintaining the cleanliness of the ICS pool water and providing the capability to add clean makeup water during normal reactor operations to offset the routine and minor loss of water inventory due to evaporation.

### **1.7.12 Fuel Pool Cooling and Clean System**

The Fuel Pool Cooling and Cleanup System (FPC) provides continuous cooling by removal of the decay heat from the spent fuel and maintains the Fuel Pool temperature below specified values. The system also maintains water level and water quality in the fuel pool, and reactor cavity pool. The FPC consists of one cooling and clean-up train provided with 100% capacity during normal operation (including pool maximum heat load).

### **1.7.13 Containment Inerting System**

The Containment Inerting System precludes the combustion of hydrogen and prevents damage to essential equipment and structures. It establishes and maintains an inert atmosphere ( $\leq 4\%$  dry-basis-percent (DB%) oxygen) within containment during plant operating modes except during shutdown for refueling/maintenance and for limited periods of time during low power operation for inspection. The system also maintains a slightly positive pressure in containment to prevent air (oxygen) in-leakage into the inerted spaces from the Reactor Building.

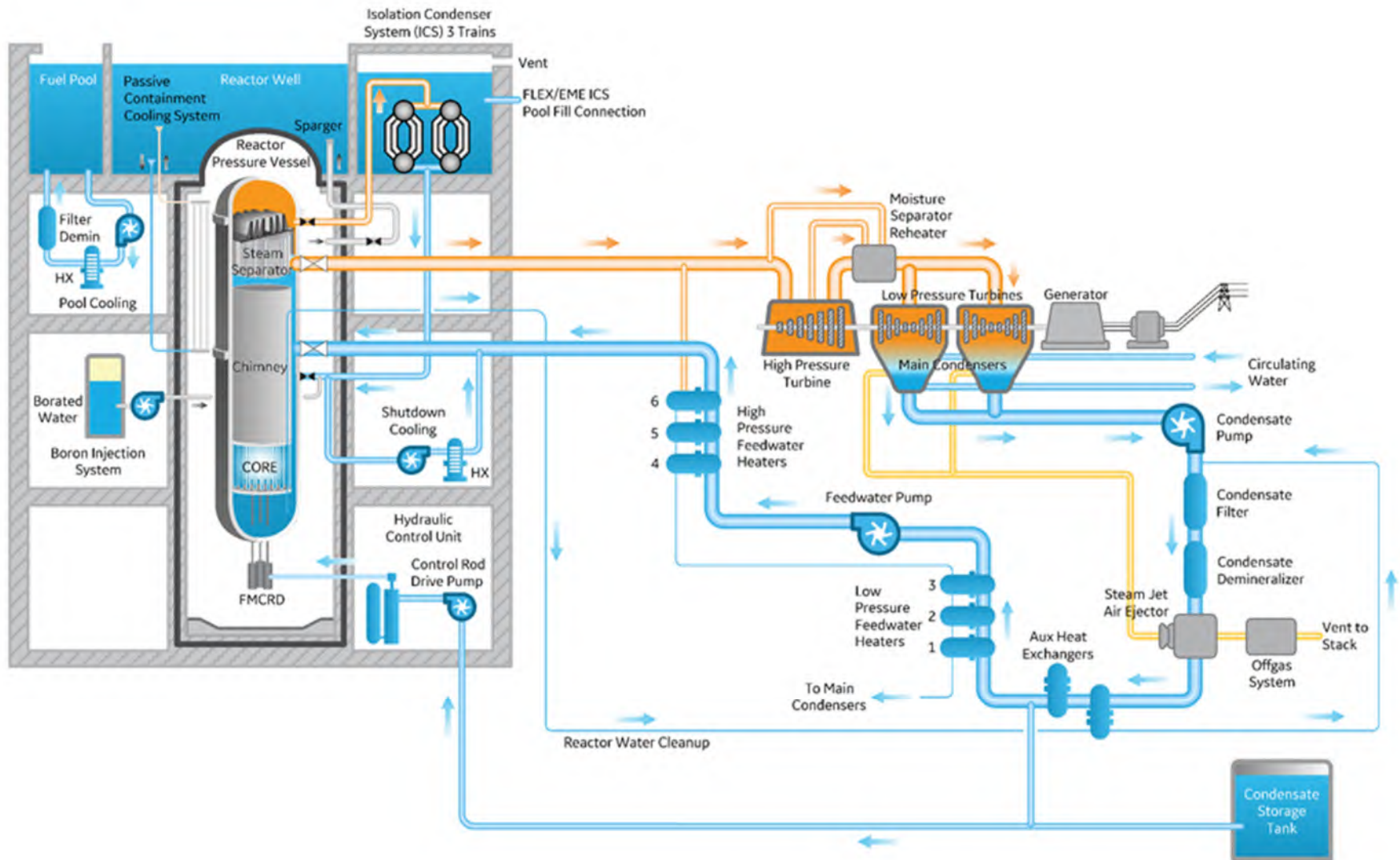


Figure 1.7-1: BWRX-300 Major Systems

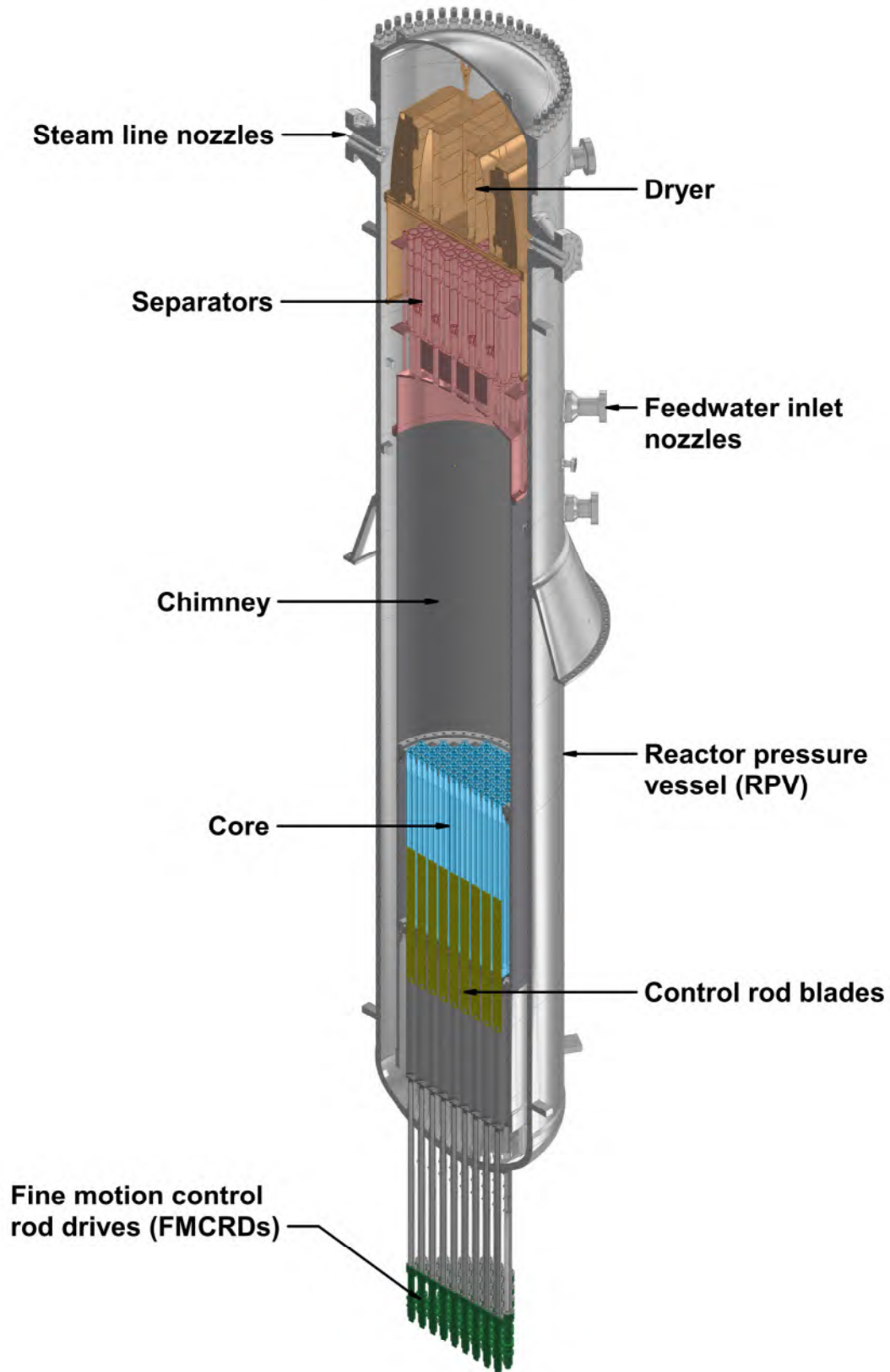
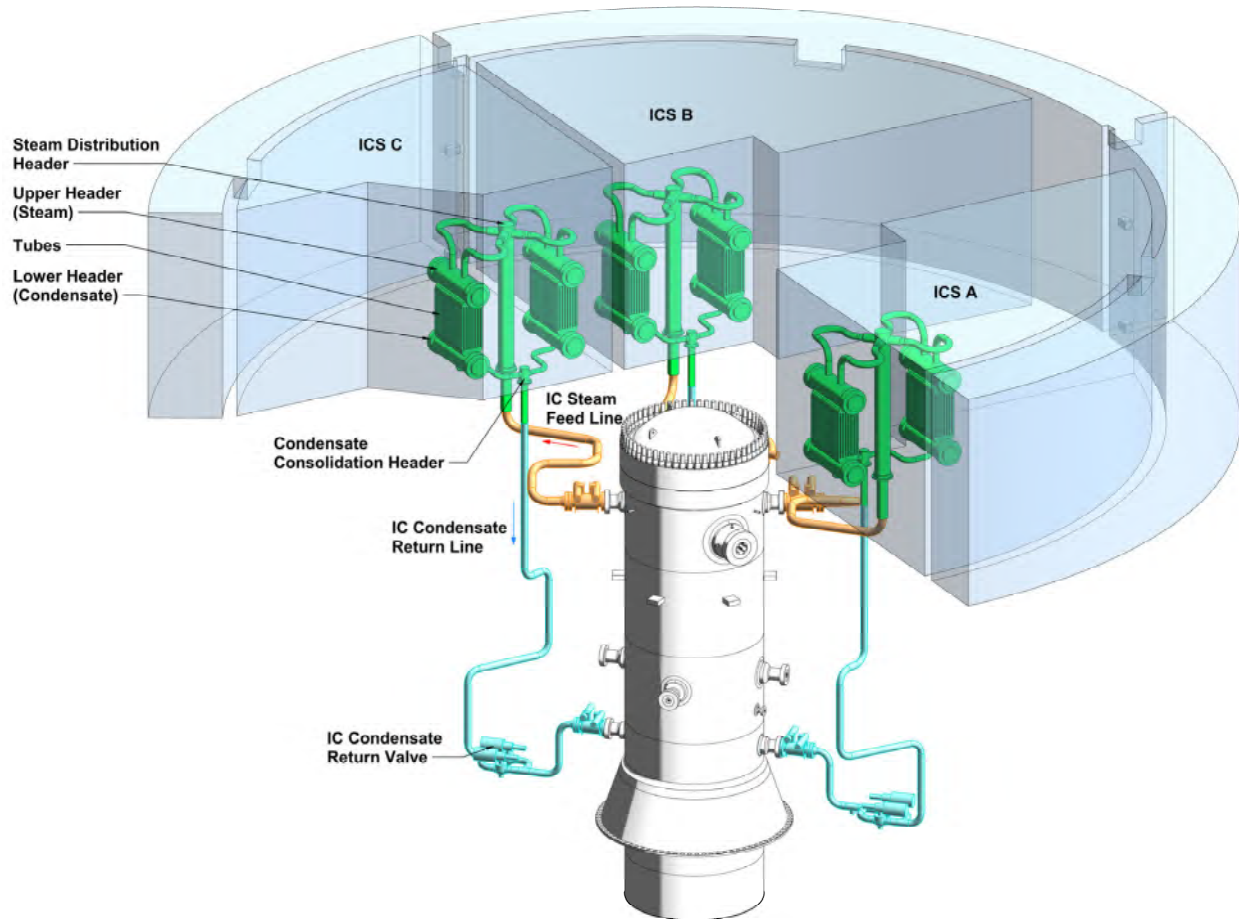


Figure 1.7-2: BWRX-300 RPV and Internals



**Figure 1.7-3: Isolation Condenser System**

### **1.8 Modes of Normal Operation of the Plant**

The normal operating modes are listed below and described in detail in Chapter 16. Chapter 15 includes a discussion of the plant design envelope, which comprises all plant states considered in the design, normal operation, Anticipated Operational Occurrence (AOO), Design Basis Accident (DBA), and Design Extension Condition (DEC).

Specific operational states and accident conditions and responses to these events for the BWRX-300 design are described in the deterministic safety analyses in Chapter 15.

The normal plant operational modes are listed below:

- Mode 1        Power Operation
- Mode 2        Startup
- Mode 3        Hot Shutdown
- Mode 4        Stable Shutdown
- Mode 5        Cold Shutdown
- Mode 6        Refueling



## **1.9 Principles of Safety Management**

The prime responsibility for safety of the facility rests with OPG which is the owner, operator, and licensee. This responsibility includes operating activities performed by OPG and the oversight of activities performed by contracted organizations, such as design, procurement construction, commissioning, and decommissioning. All activities performed, either directly by OPG or indirectly under OPG's oversight, are controlled in accordance with OPG's N-CHAR-AS-0002, "Nuclear Management System" (Reference 1.9-1), which is the top management system of the facility. The system is implemented by a series of program documents which in turn define the required implementing procedures and standards. Chapter 17 of this PSAR provides details regarding the management for safety, including the different management processes aimed at ensuring safety is given the highest priority, the specific elements of the management system, quality management, and the nuclear safety culture framework.

The management system promotes safety culture by committing workers to adhere to its implementing practices that contribute to the excellence in worker performance, supporting workers in carrying out their tasks safely and successfully, and monitoring to improve the culture. The organizational structure implements the programs that make up the OPG management system with the Chief Nuclear Officer accountable for its implementation and effectiveness.

A structured operating organization is established having defined authorities, managerial responsibilities, interfaces between organizations and policies for use of contracted resources such that safety is the overriding priority. An organizational approach is taken that assures the required capabilities and qualifications necessary to always maintain nuclear safety and the integrity of the safety case. This includes maintaining sufficient capability within the operating organization to effectively manage the design and licensing basis of the facility and preventing the over-reliance on contractors. Additionally, the operating organization ensures changes having any potential impact on the safety of the public and workers, the environment, and Canada's international obligations are thoroughly assessed and demonstrated to be acceptable throughout the life of the facility.

### **1.9.1 References**

1.9-1 N-CHAR-AS-0002, "Nuclear Management System," Ontario Power Generation.

#### **1.10 Additional Supporting or Complementary Documents to the Safety Analysis Report**

Table 1.10-1 lists all GE, GNF and GEH topical reports that are incorporated by reference in this PSAR document. These reports impose requirements on the BWRX-300 design.

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 1.10-1: Topical Reports Incorporated by Reference**

Report No.	Title	Section No.
EPRI NP-2660	Fire Tests in Ventilated Rooms-Extinguishment of Fires in Grouped Cable Trays, Electric Power Research Institute, Palo Alto, California, 1982	9A
EPRI NP-5479	Application Guide for Check Valves in Nuclear Power Plants, 1993	10.2
EPRI TR-102293	Guidelines for Determining Design Basis Ground Motions, Electric Power Research Institute, Palo Alto, California, Vol. 1-5, 1993	3.3
EPRI TR-103959	Methodology for Developing Seismic Fragilities, Electric Power Research Institute Palo Alto, California, 1994	3.5
EPRI TR-1002988	Seismic Fragility Application Guide, Electric Power Research Institute, Palo Alto, California, 2002	3.5
EPRI TR-1006756	Fire Protection Equipment Surveillance Optimization and Maintenance Guide, 2018	
EPRI TR-1011989 USNRC NUREG/CR-6850	Fire PRA Methodology for Nuclear Power Facilities, 2010	15.6
EPRI TR-1019200	Seismic Fragility Application Guide Update, Electric Power Research Institute, Palo Alto, California, 2009	3.5
EPRI TR-3002002623	BWR Vessel and Internals Project, Volumes 1&2: BWR Water Chemistry Guidelines – Mandatory, Needed, and Good Practice Guidance, Palo Alto, California, 2014	5.2
EPRI TR-3002012994	Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments, 2018	3.5
NEDO-10871	General Electric Company, “Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms,” March 1973	11.1
NEDO-10958-A NEDE-10958P-A	General Electric Thermal Analysis Basis Data, Correlation and Design Application, January 1977	4.4
NEDO-11209-A	GE Hitachi Nuclear Energy, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” Class I (Non-proprietary), NEDO-11209-A, Revision 16, December 2020	9A.6, 13.3, 17.2
NEDE-24011-P-A-31	GNF General Electric Standard Application for Reactor Fuel, November 2020, U.S. Supplement	4.4, 4.7, 15.3
NEDC-32082P	BWR Steady-State Thermal Hydraulic Methodology (ISCOR), Revision 0, August 1992	4.3, 15.5
NEDE-32176P	Licensing Topical Report TRACG Model Description, Revision 4, January 2008	4.3
NEDO-32177 NEDE-32177P	“TRACG Qualification,” Class I (Non-proprietary), Revision 3, August 2007	4.4, 4.7, 15.5

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Report No.	Title	Section No.
NEDO-32601-A NEDC-32601P-A	Methodology and Uncertainties for Safety Limit MCPR Evaluations, August 1999	4.4
NEDO-32708	Radiological Accident Evaluation – The CONAC04A Code, August 1997	15.5
NEDC-32725P	TRACG Qualification for SBWR, Revision 1, Vol. 1 and 2, August 2002	4.7, 15.5
NEDC-33080 NEDC-33080P	TRACG Qualification for ESBWR, Revision 1, May 2005	4.7, 15.5
NEDO-33083-A NEDC-33083P-A	“TRACG Application for ESBWR,” Revision 1, September 2010	15.5
NEDO-33083 Supplement 1-A NEDE-33083 Supplement 1P-A	TRACG Application for ESBWR Stability Analysis, Revision 2, September 2010	4.7
NEDC-33139P-A	Global Nuclear Fuel, Cladding Creep Collapse, July 2005	4.2
NEDC-33256P-A	GE Hitachi Nuclear Energy, Licensing Topical Report – The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1-Technical Bases, Revision 2, October 2021	4.2
NEDC-33257P-A	Licensing Topical Report – The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2-Qualification, Revision 2, October 2021	4.2
NEDC-33258P-A	Licensing Topical Report – The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3-Application Methodology, Revision 2, October 2021	4.2
NEDC-33270P	GE Nuclear Energy, GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTARII), Revision 11, August 2020	4.2
NEDE-33284 Supplement 1P-A	GE Hitachi Nuclear Energy, “Marathon-Ultra Control Rod Assembly”, “Global Nuclear Fuels, Fuel Bundle Designs,” Class III (Proprietary), Revision 1, March 2012	4.2
NEDO-33292 NEDC-33292P	Global Nuclear Fuel, GEXL17 Correlation for GNF2 Fuel, Revision 3	4.4
NEDO-33798-A NEDE-33798P-A	Global Nuclear Fuel, “Application of NSF to GNF Fuel Design,” Revision 1, September 2015	4.2
NEDC-33840P-A	Global Nuclear Fuel, Class II (Internal), “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” Rev 1, August 2017.	4.2
NEDC-33910P-A	GEH Licensing Topical Report, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” Revision 2, June 20201	5.1, 6.2

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Report No.	Title	Section No.
NEDE-33911P-A	GE Hitachi Nuclear Energy, BWRX-300 Containment Performance, Revision 3, January 2022	5.1, 6.2, 6.3
NEDO-33914-A	Licensing Topical Report, BWRX-300 Advanced Civil Construction and Design Approach, Revision 2, June 2022	21.3
NEDO-33914 NEDE-33914P	Licensing Topical Report, BWRX-300 Advanced Civil Construction and Design Approach, Revision 0, January 2021	3.2, 3.3, 3.5, 9B
NEDO-33922 NEDC-33922P	GEH Licensing Topical Report, "BWRX-300 Containment Evaluation Method," Revision 3, June 2022	5.1, 6.3, 15.5
NEDC-33939	"Steady State Nuclear Methods TGBLA06/PANAC11 Application Methodology For BWRX-300", August 2022	4.3
NEDC-33940P NEDO-33940	GNF2 Fuel Assembly Mechanical Design Report for BWRX-300, September 2022	4.2
NEDC33941P NEDO-33941	Class II (Proprietary), "GNF2 Fuel Rod Thermal Mechanical Design Report," R1 August 2022	4.2
NEDC-33946P	BWRX-300 Darlington New Nuclear Project (DNNP) Probabilistic Safety Assessment Methodology, Revision 0, September 2022.	15.6
NEDC-33974P	BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report	3.2
NEDC-33976	GNF2 Pressure Drop Calculations, Rev 0, August ,2022	4.4
NEDC-33977P	BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Design Description, Qualification and BWR Fuel Licensing, Revision 0, August 2022	4.2
NEDC-33982P	BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan	18.1, 18.2, 18.3
NEDC-33985P	Nuclear Design Report for BWRX-300 Equilibrium 12 Month Cycle, Revision 1, July 2022	4.3
NEDCC-33987	BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application for BWRX-300, Rev. 0, September, 2022	4.3
NEDC-33992P-A	BWRX-300 Containment Evaluation Method, Revision 3, June 2022	6.2

### **1.11 Conformance with Applicable Regulations, Codes and Standards**

The Nuclear Safety and Control Act (NSCA) establishes the Canadian Nuclear Safety Commission (CNSC) and provides the CNSC with the authority to regulate the development, production and use of nuclear energy and the production, possession and use of nuclear substances, prescribed equipment, and prescribed information in Canada. Class I Nuclear Facilities Regulations (SOR/2000-204) are applicable to nuclear fission reactors that includes the BWRX-300 at the DNNP facility. It provides the requirements for the different types of applications for Class I nuclear facilities.

The CNSC's regulatory framework, shown in Figure 1.11-1 below, consists of laws passed by Parliament that govern the regulation of Canada's nuclear industry, and regulations, licences, and documents that the CNSC uses to regulate the industry. The CNSC publishes regulatory documents (REGDOCs) that are instruments that clarify, resulting in consistent implementation of, regulatory requirements and expectations. Regulatory requirements and expectations are further supported by codes and standards published by domestic and international agencies.



**Figure 1.11-1: CNSC Regulatory Framework**

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Laws, regulations, codes, and standards are one of numerous sources of design requirements that the design of a Nuclear Power Plant (NPP) must satisfy. The RCS applicable to the licensing basis and the design basis of SSCs within the Protected Area are managed in accordance with the GEH BWRX-300 requirement management plan that is depicted in Figure 1.11-2 below.

The identification and implementation of design requirements, which includes applicable RCS, is an iterative process that evolves with the maturity of the design and is managed throughout the design lifecycle. To align with the requirements management plan, the following levels of RCS are defined in support of the licensing process and the PSAR development:

- Source Level RCS (licensing basis)
- Plant Level RCS (design basis)
- System Level RCS (design basis)
- Component Level RCS (design basis)

Source Level RCS are those applicable jurisdictional requirements that establish part of the licensing basis of the SSCs within the Protected Area.

Plant System, and Component Level RCS are those that govern the design of the facility that establishes the design basis. As described above, the identification and implementation of the relevant level RCS is dependent on the maturity of the design and may not have been implemented from the onset of the design. For the purposes of the LTC and the PSAR, Plant Level RCS is commensurate with the state of design progression.

The above levels recognize that the identification and implementation of RCS is a managed and iterative process. As indicated in Figure 1.11-2, decisions that support the identification, selection, and implementation of design requirements, including RCS, at the various levels are documented and controlled. The repository used for the management of applicable RCS, including supporting decisions, throughout the design lifecycle of the facility is the requirements management tool.



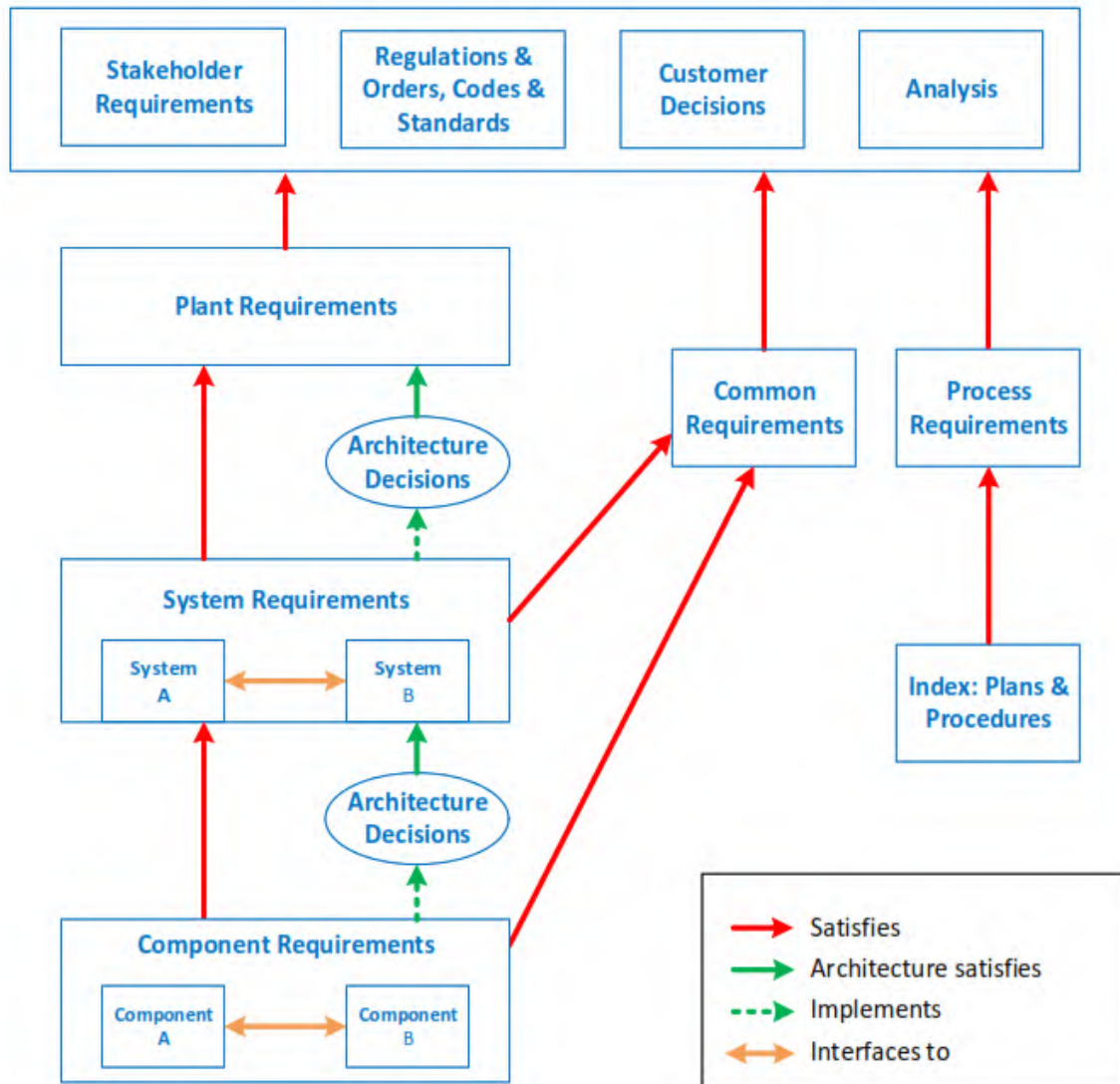


Figure 1.11-2: GEH Requirements Hierarchy

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

The Source Level licensing basis RCS that are applicable to the LTC of the facility along with the methodology used in its development are documented in NK054-REP-01210-00137, "DNNP License to Construct Regulatory Documents, Codes & Standards," (Reference 1.11-1).

The design basis RCS that governs the design of the facility, including the BWRX-300 reactor, are documented in Appendix B. These RCS contain design related requirements applicable to the facility. RCS reference throughout the PSAR, not listed in Appendix B, are used for guidance only.

Table B1.11-1 includes the list of applicable Source Level design basis REGDOCs that originates from Source Level licensing basis REGDOCs documented in Reference 1.11-1 that are screened to eliminate those not applicable to the design of Power Block.

Table B1.11-2 includes the list of applicable Plant Level design basis codes and standards that originates from Source Level licensing basis codes and standards documented in Reference 1.11-1 that are screened to eliminate those not applicable to the design of the facility.

Table B1.11-3 includes the list of Plant Level US regulatory codes that are applicable to the facility. The method used to develop the lists followed this general process:

The strategy used to evaluate codes and standards for their applicability, sufficiency, and adequacy for the OPG BWRX-300 design is based on the licensing basis codes and standard provided by OPG.

1. The licensing basis list is evaluated to determine which of the codes and standards forms the design basis.
2. Subsequent review of the GEH documentation is completed to determine the remaining codes and standards to develop the plant level design basis.
3. The system level list is next developed with the codes and standards from the system design documentation.
4. Each code and standard are reviewed for applicability with the responsible design engineer.

#### **1.11.1 References**

1.11-1 NK054-REP-01210-00137, "DNNP License to Construct Regulatory Documents, Codes & Standards," Ontario Power Generation.

## 1.12 Appendix A – Darlington Nuclear Site and DNNP General Arrangement Drawings

This Appendix A includes the following figures:

Figure No.	Description
A1.1-1	Darlington Nuclear Site Regional Location
A1.1-2	Darlington Nuclear Site (DNNP Proximity to DNGS)
A1.4-1	DNNP BWRX-300 Facility Site Layout
A1.4-2	DNNP Switchyard Site Plan
A1.5-1	BWRX-300 Power Block Plan View at Elevation 0

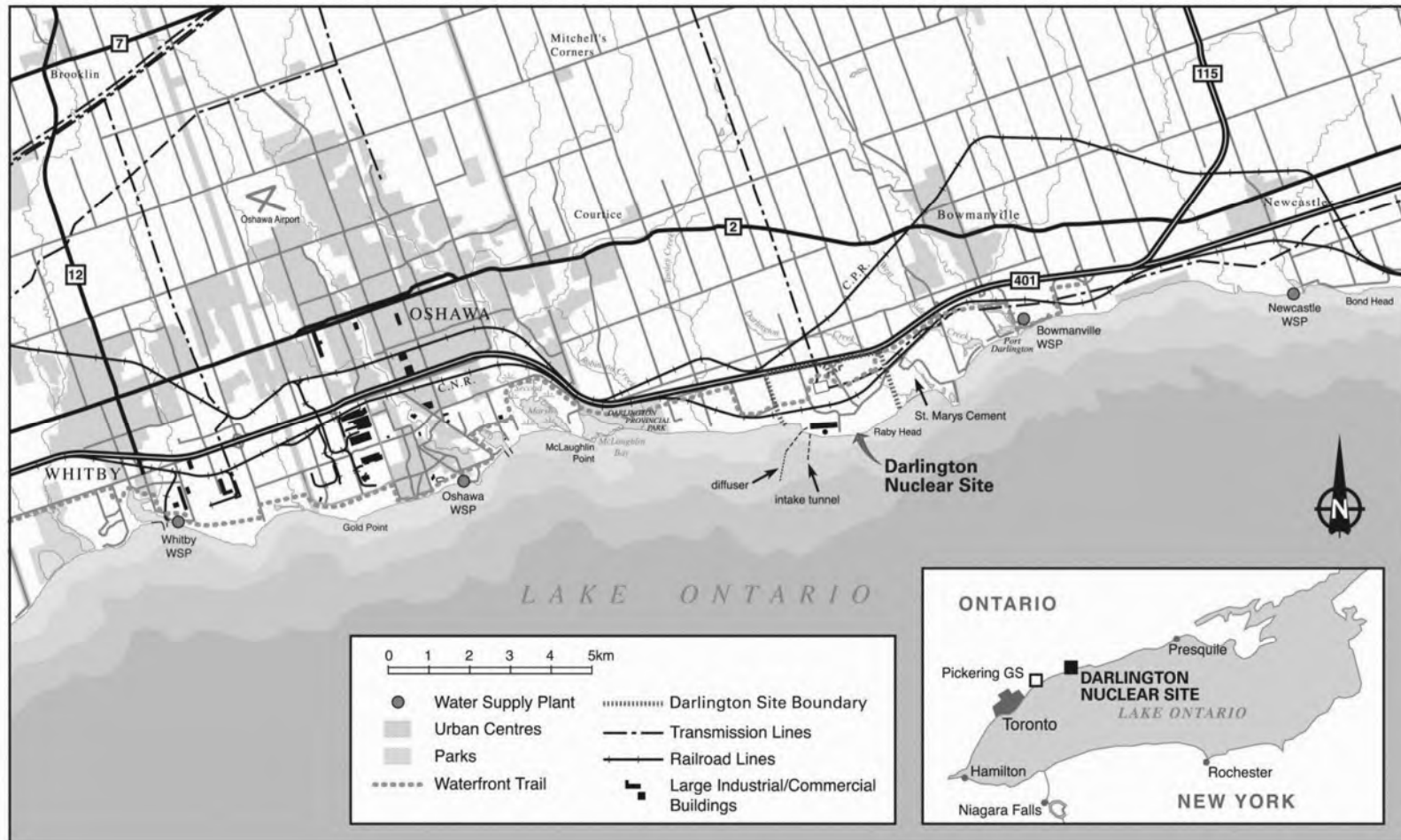


Figure A1.1-1: Darlington Nuclear Site Regional Location

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

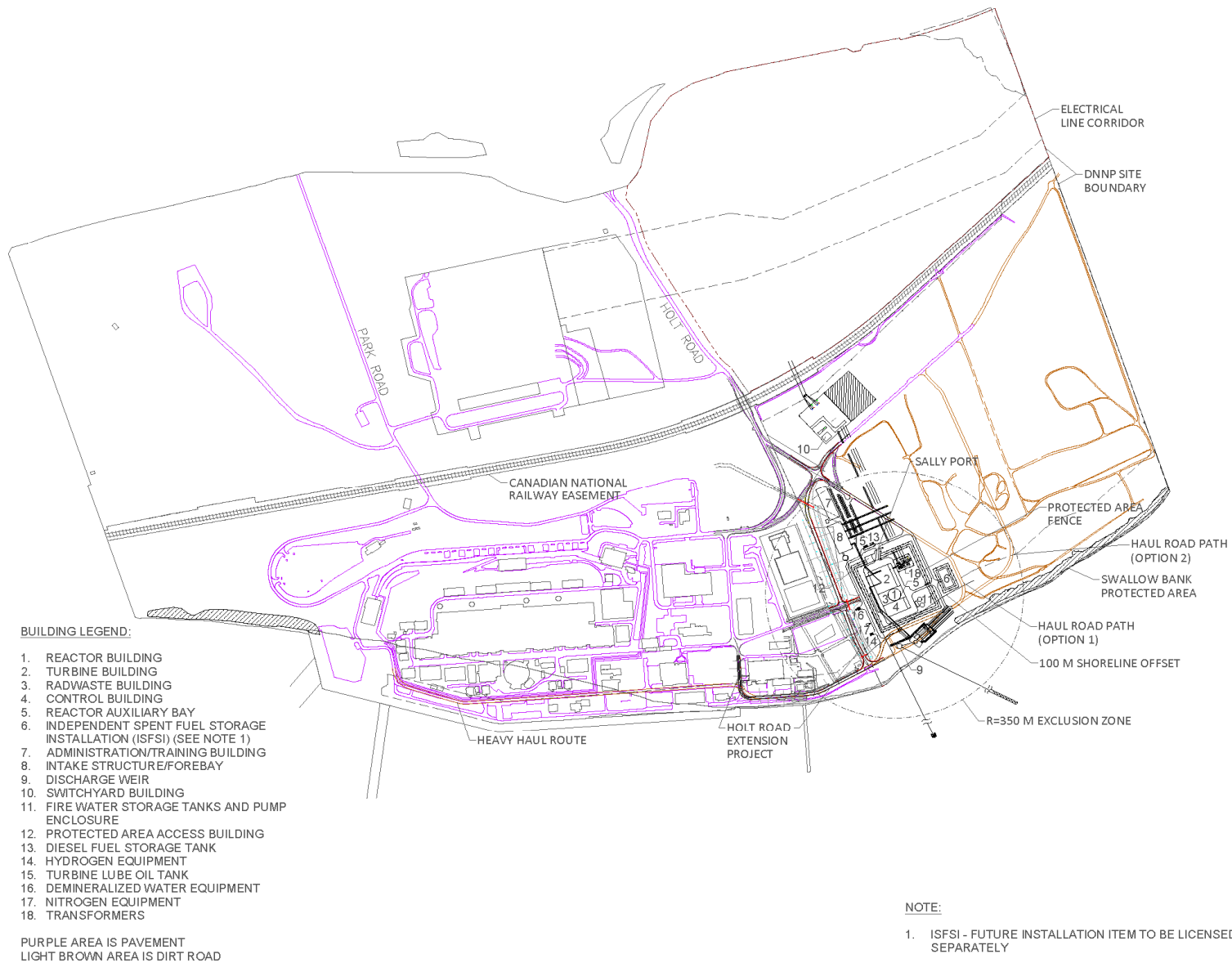
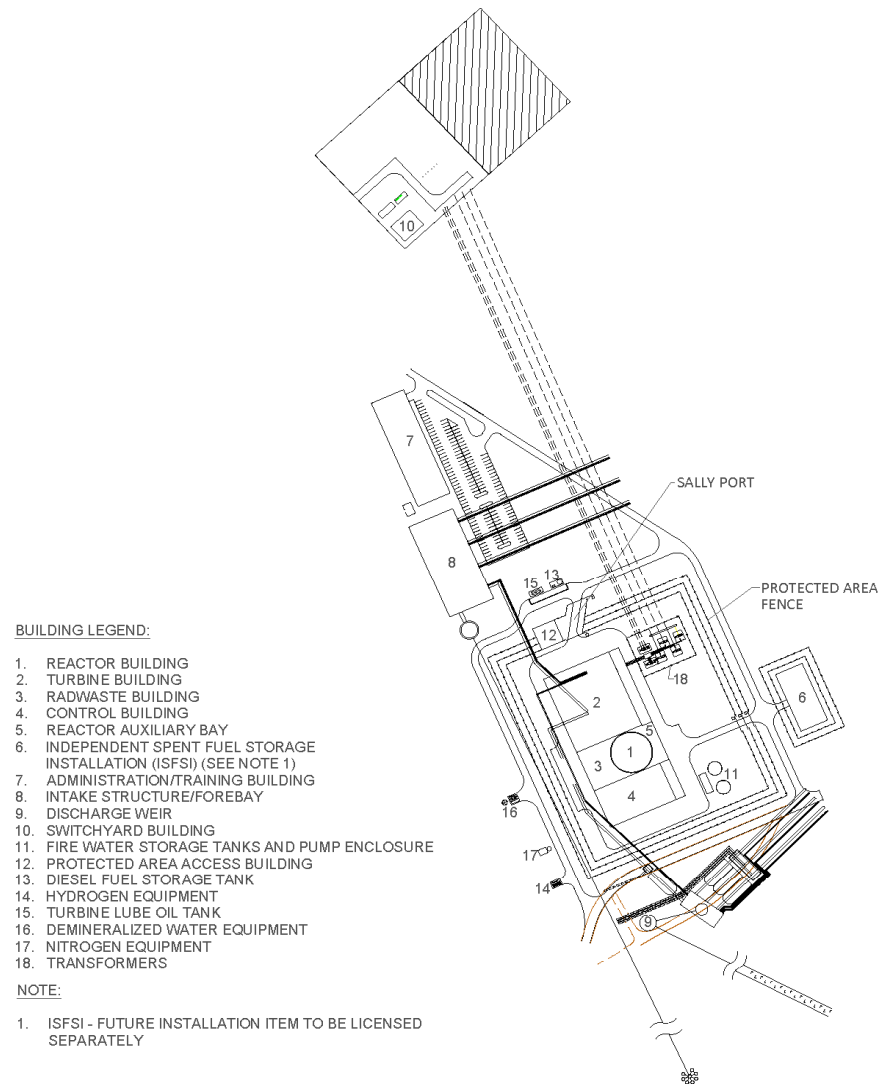


Figure A1.1-2: Darlington Nuclear Site (DNNP Proximity to DNGS)

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure A1.4-1: DNNP BWRX-300 Facility Site Layout**

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

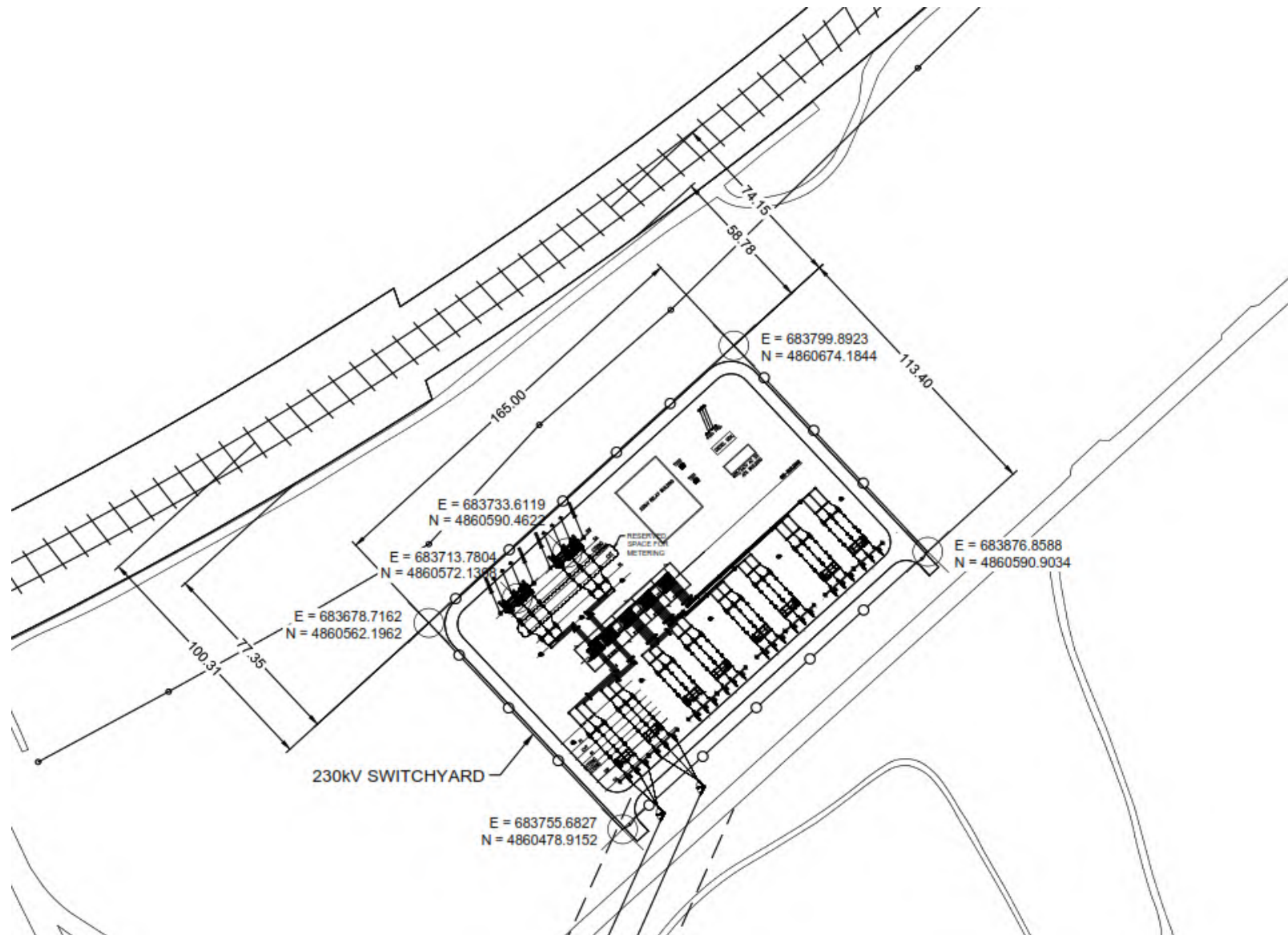


Figure A1.4-2: DNNP Switchyard Site Plan



NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

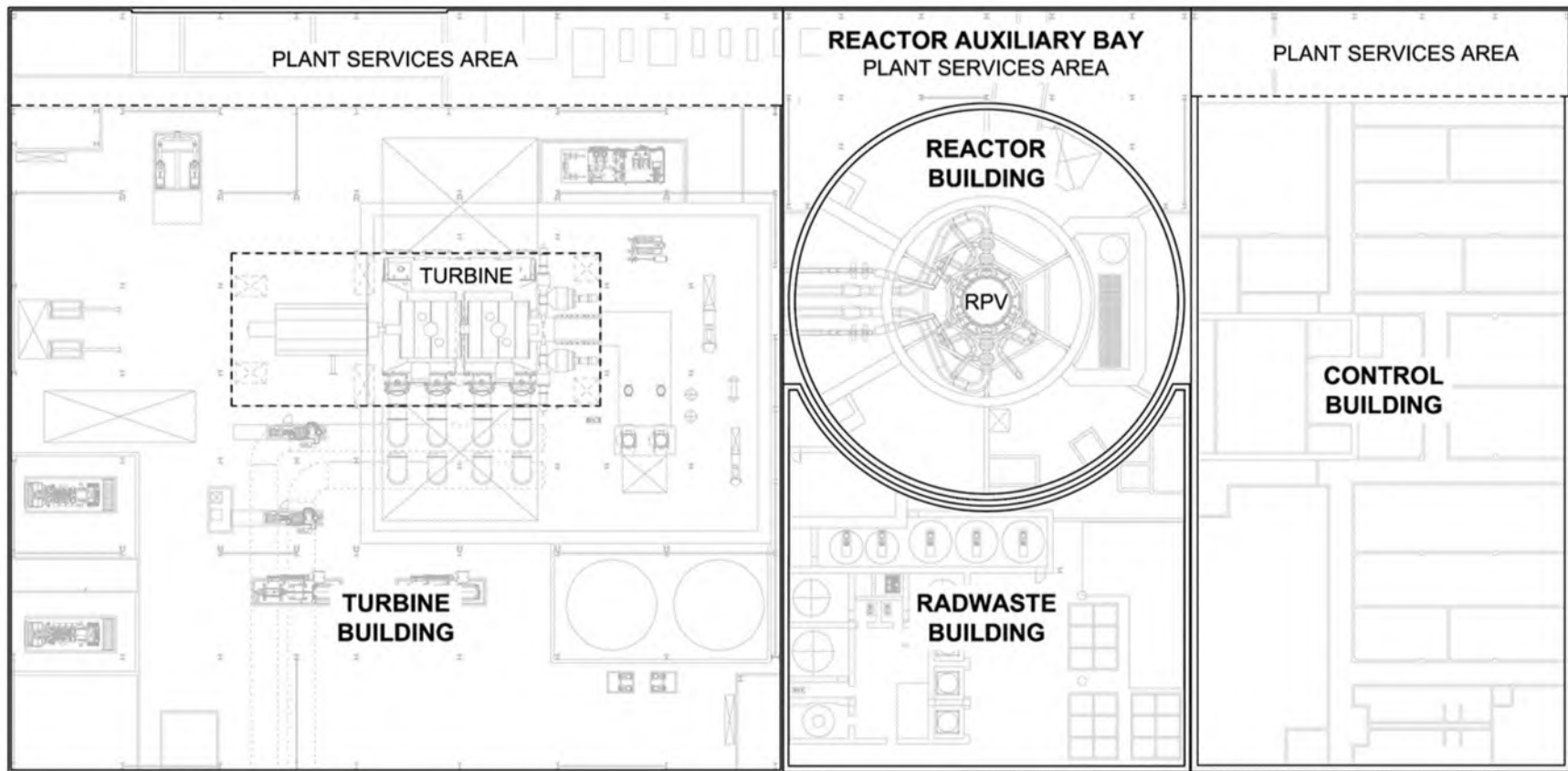


Figure A1.5-1: BWRX-300 Power Block Plan View at Elevation 0



**1.13 Appendix B – Tables of Design Basis REGDOCs, Codes, and Standards**

This Appendix includes the following Tables:

<b>Table No.</b>	<b>Description</b>
B1.11-1	List of Plant Level Design Basis Regulatory Documents
B1.11-2	List of Plant Level Design Basis Codes and Standards
B1.11-3	List of Plant Level Design Basis US Regulatory Documents

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table B1.11-1: List of Plant Level Design Basis Regulatory Documents**

Document Number	Document Title	Doc Effective Date/Version
CNSC REGDOC -2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2022
CNSC REGDOC -2.4.3	Nuclear Criticality Safety	2020
CNSC REGDOC 2.7.1	Radiation Protection	2021
CNSC REGDOC 2.9.1	Environmental Protection: Environmental Principles, Assessments and Protection Measures	2021
CNSC REGDOC-2.12.1	High Security Facilities, Volume I: Nuclear Response Force	2018
CNSC REGDOC-2.12.1	High-Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices	2018
CNSC REGDOC-2.13.1	Safeguards and Nuclear Material Accountancy	2018
CNSC REGDOC-2.3.2	Accident Management	2015
CNSC REGDOC-2.4.1	Deterministic Safety Analysis	2014
CNSC REGDOC-2.5.1	General Design Considerations: Human Factors	2019
CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2014
CNSC REGDOC-2.6.1	Reliability Programs for Nuclear Power Plants	2017
CNSC REGDOC-2.6.3	Aging Management	2014

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table B1.11-2: List of Plant Level Design Basis Codes and Standards**

<b>Document Number</b>	<b>Document Title</b>	<b>Doc Effective Date/Version</b>
ACI 350.3	Seismic Design of Liquid-Containing Concrete Structures	2006
ANSI/AISC N690	Specification for the Design, Fabrication and Erection of Steel Safety Related Structures for Nuclear Facilities	2018
API 620	Design and Construction of Large, Welded, Low-Pressure Storage Tanks	12th edition
API 650	Welded Tanks for Oil Storage	13th edition
ASCE/SEI 4	Seismic Analysis of Safety-Related Nuclear Structures	2016
ASCE/SEI 43	Seismic Design Criteria or Structures, Systems, and Components in Nuclear Facilities	2019
ASCE/SEI 7	Minimum Design Loads for Buildings and Other Structures	2016
ASME B31.1	Power Piping	2020
ASME B31.3	Process Piping	2020
ASME BPVC Section II	Materials	2021
ASME BPVC Section III	Rules for Construction of Nuclear Facility Components	2021
ASME BPVC Section IX	Welding, Brazing, and Fusing	2021
ASME BPVC Section V	Non-Destructive Examination	2021
ASME BPVC Section VIII	Rules for Construction of Pressure Vessels	2021
ASME BPVC Section XI	Rules for Inservice Inspection of Nuclear Power Plant Components	2021
ASME/ANS RA-Sb-2013	Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications	2013
CSA A23.1	Concrete Materials And Methods Of Concrete Construction	2019
CSA A23.2	Test Methods And Standard Practices For Concrete	2019
CSA A23.3	Design of Concrete Structures	2019

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Document Number	Document Title	Doc Effective Date/Version
CSA C22.1	Canadian Electrical Code, Part 1 Safety Standard for Electrical Installation	2021
CSA C22.2	Canadian Electrical, Part 2 General Requirement	2021
CSA N1600	General requirements for nuclear emergency management programs	2021
CSA N285.0/N285.6*	General Requirements For Pressure-Retaining Systems And Components In CANDU Nuclear Power Plants/Material Standards For Reactor Components For CANDU Nuclear Power Plant	2017
CSA N288.4	Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	2019
CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	2011
CSA N288.7	Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills	2020
CSA N289.1	General requirements for seismic design and qualification of nuclear power plants	2018
CSA N289.2	Ground motion determination for seismic qualification of nuclear power plants	2021
CSA N289.3	Design procedures for seismic qualification of nuclear power plants	2020
CSA N289.4	Testing procedures for seismic qualification of nuclear power plant structures, systems, and components	2012 (R2017)
CSA N289.5	Seismic instrumentation requirements for nuclear power plants and	2012
CSA N290.7	Cyber Security for nuclear power plants and small reactor facilities	2021
CSA N290.11	Reactor heat removal capability during outage of nuclear power plants	2021
CSA N290.12	Human Factors In Design For Nuclear Power Plants	2014 (R2019)
CSA N290.13	Environmental qualification of equipment for nuclear power plants	2018

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Document Number	Document Title	Doc Effective Date/Version
CSA N290.14	Qualification of Digital Hardware and Software for Use in Instrumentation and Control Applications for Nuclear Power Plants	2015
CSA N291	Requirements for Safety-Related Structures for Nuclear Power Plants	2019
CSA N293S1	Supplement No. 1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)	2021
CSA S16	Design of Steel Structures	2019
CSA W59	Welded Steel Construction	2018
IEC 60034-1	Rotating Electrical Machines – Part 1: Ratings and Performance	2022
IEC 60099-5	Surge Arresters – Part 5: Selection and Application Recommendations – Edition 3.0	2018
IEC 60137	Insulated Bushings for Alternating Voltages Above 1000 V	2017
IEC 60152	Designation of Phase Differences by Hour Numbers in Three Phase AC Systems	2021
IEC 60255-1	Measuring Relays and Protection Equipment – Part 1: Common Requirements	2009
IEC 60772	Nuclear Power Plants - Instrumentation Systems Important to Safety - Electrical Penetration Assemblies in Containment Structures	2018
IEC 60880	Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions	2006
IEC 60987	Nuclear Power Plants - Instrumentation and Control Important to Safety - Hardware Requirements	2021
IEC 61000-6-2	Immunity standard for industrial environments	2019
IEC 61500	Network Communication	2018
IEC 61513	Instrumentation and Control Important to Safety – General Requirements for Systems	2011
IEC 62040-1	Uninterruptible Power Systems (UPS) – Part 1: Safety Requirements	2021
IEC 62041	Transformers, power supplies, reactors, and similar products – EMC requirements	2017

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

Document Number	Document Title	Doc Effective Date/Version
IEC 62138	Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category B or C Functions	2018
IEC 62271-103	High-voltage Switchgear and Control gear – Part 103: Alternating Current Switches for Rated Voltages Above 1 kV Up To and Including 52 kV	2021
IEC 62566	Nuclear Power Plants – Instrumentation and Control Important to Safety – Development of HDL-Programmed Integrated Circuits for Systems Performing Category A Functions	2012
IEC 62566-2	Nuclear Power Plants – Instrumentation and Control Important to Safety – Development of HDL-Programmed Integrated Circuits – Part 2: HDL Programmed Integrated Circuits for Systems Performing Category B or C Functions	2020
IEC 62859	Nuclear power plants – Instrumentation and control systems – Requirements for coordinating safety and cybersecurity	2016
IEC 63147	Criteria for accident monitoring instrumentation for nuclear power generating stations	2017
IEEE Std 80	Guide for Safety in AC Substation Grounding	2019
IEEE Std 81	Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Grounding System	2012
IEEE Std 384	Standard Criteria for Independence of Class 1E Equipment and Circuits	2018
NFPA 10	Standard for Portable Fire Extinguishers	2018
NFPA 13	Standard for the Installation of Sprinkler Systems	2022
NFPA 15	Standard for Water Spray Fixed Systems for Fire Protection	2007
NRCC NBC	National Building Code	2020
NRCC NFC	National Fire Code	2020

\*Pressure boundary and jurisdictional requirements call for the use of CSA N285 supplemented by ASME BPVC and US Regulatory guides. Pressure boundary requirements appropriate for the BWRX-300 are documented per NK054-REP-01210-00137 (Reference 1.11-1).

NEDO-33950 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table B1.11-3: List of Plant Level Design Basis US Regulatory Documents**

Document Number	Document Title	Doc Effective Date/Version
US NRC 10CFR50 Appendix A	General Design Criteria for Nuclear Power Plants	N/A
US NRC 10CFR50 Appendix J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors	N/A
US NRC 10CFR50.55a	Codes and Standards	N/A
US NRC RG 1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2001
US NRC RG 1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	2019
US NRC RG 1.243	Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments	2021
US NRC RG 1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	2021
US NRC RG 1.61	Damping Values for Seismic Design of Nuclear Power Plants	2007
US NRC RG 1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	2019
US NRC RG 5.71	Cyber Security Programs for Nuclear Facilities	2010



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33951

Revision 2

March 7, 2023

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 2  
Site Characteristics**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	Section 2.1.2 Section 2.2.3 Section 2.5.1 Section 2.6.4 Section 2.6.5 Section 2.6.8 Section 2.6.9 Section 2.7.1 Section 2.7.2 Section 2.7.3 Section 2.8.2 Section 2.11.4 Section 2.12.1 Section 2.12.5 Section 2.12.8	Incorporated corrections per customer acceptance review
2	All	Edited to improve readability, streamline the text, and ensure consistency across all sections of Chapter 2
	All	Several paragraphs are deleted for they became irrelevant, outdated, or obsolete due to the incorporation of recent (2022 and 2023) information generated in works involving DNNP site-specific investigations, analyses, and assessments.
	All Summary Tables	The tables at the beginning of each section are updated to reflect the edited and added contents of corresponding texts in that section.
	All Other Tables	Other tables are updated or replaced with inputs from new characteristics and parameters generated in the site-specific studies completed in 2022 and 2023.
	All Figures	Updated or replaced to reflect the new information resulted from several 2022 and 2023 assessments, investigations, and analyses
	Acronym List	Updated to include added acronyms

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
	Section 2.0	Chapters 7, 19 and 20 added to the list of key chapters, and edits involving details are made to previously listed chapters
	Section 2.1.1	Paragraphs added on Site Topography regarding the different grade elevations at and around the Darlington Nuclear site
	Section 2.1.1	Edited to incorporate information in Reference 2.1-7
	Section 2.1.2	Edited to reflect current contents of Chapter 9B, and to incorporate the information in the 2022 Environmental Impact Assessment in Reference 2.1-4
	Section 2.1.2.1	Edited to incorporate information in the 2022 References 2.1-4, 2.1-5, and 2.1-6
	Section 2.1.2.3	A new bullet added to reflect information on the heavy haul routes described in Reference 2.1-4
	Section 2.1.2.4	Added bullets number 6 And 7 regarding not using the cooling towers and combing the primary and secondary heat transport systems
	Section 2.1.10	Added seven new References 2.1-4 to 2.1-9
	Section 2.2.2	Added a paragraph on the 2022 DNNP Hazard Analysis Methodology (Reference 2.2-10)
	Section 2.2.3.2	Edited to incorporate information in the 2022 assessments reported in Reference 2.2-11 and Reference 2.2-12
	Section 2.2.5.2	Edited to incorporate information in the 2022 PNGS re-assessment documented in Reference 2.2-13
	Section 2.2.11	Three references added: 2.2-13, 2.2-14 and 2.2-15
	Section 2.4.2	Paragraphs added to reflect information in the 2022 EIS in Reference 2.4-2
	Section 2.4.3	Two references added: the 2022 Reference 2.4-2 and the 2009 Reference 2.4-3
	Section 2.5.2.1	Edited and updated to incorporate information in the 2022 Flood Hazard Assessment documented in Reference 2.5-18
	Section 2.5.3 and associated subsections	Edited and updated to incorporate information in the 2022 Reference 2.5-18, Reference 2.5-19
	Section 2.5.3.1	Deleted DNGS information that became irrelevant

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
	Section 2.5.3.3	A new Table 2.5-2 is added
	Section 2.5.3.4	Deleted DNGS information that became irrelevant and edited to incorporate information in the 2022 Reference 2.5-18
	Section 2.5.4	Edited and updated to incorporate information in the 2022 Reference 2.5-18 and the 2023 Climate Change Impact Strategy documented in Reference 2.5-20
	Section 2.5.5 and associated subsections	Edited and updated to incorporate information in the 2022 References 2.5-18 and 2.5-21
	Section 2.5.6 and associated subsections	Edited and updated to incorporate information in the 2022 Reference 2.5-18
	Section 2.5.7	Added four new References: 2.5-18 and 2.5-21
	Section 2.6.2	Edited and updated to incorporate information in the 2022 Flood Hazard Assessment in Reference 2.6-17
	Section 2.6-4	Edited and updated to incorporate information in the 2022 Reference 2.6-17
	Section 2.6-5	Edited and updated to incorporate information in the 2022 Wind Gust Analysis in Reference 2.6-14
	Section 2.6-5	Added new Table 2.6-3 and Table 2.6-4
	Section 2.6.9	Edited and updated to incorporate information in the 2022 Winter PMP Validation in (Reference 2.6-15)
	Section 2.6.9	Added new Table 2.6-7
	Section 2.6.12	Edited to incorporate information in the 2023 Climate Change Impact Strategy in Reference 2.6-19
	Section 2.6.13	Added six new References: 2.6-14, 2.6-15, 2.6-16 and 2.5-18
	Section 2.7	<p>The entire Section 2.7 is re-configured to incorporate new information documented in:</p> <ol style="list-style-type: none"> <li>1. The 2023 DNNP Foundation Interface Analysis (FIA) Report (Reference 2.7-38).</li> <li>2. The 2022 DNNP geotechnical investigations and test results, Phase-1 Power Block (Reference 2.7-39)</li> <li>3. The 2023 offshore geotechnical investigations (Reference 2.7-40)</li> </ol>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
		<p>4. The 2022 DNNP-specific Probabilistic Seismic Hazard Assessment (PSHA) (Reference 2.7-41)</p> <p>5. The 2022 DNNP seismically-induced soil liquefaction assessment (Reference 2.7-42)</p> <p>Added referencing to the 2022 and 2023 completed DNNP/BWRX-300 investigations, analyses, and assessments.</p>
	Section 2.7.1	Deleted irrelevant DNGS information and outdated information
	Section 2.7.2.4	Added information based on the 2023 offshore investigations (Reference 2.7-40), and deleted outdated information
	Section 2.7.3.1	Updated relevant figures, and added information based on the 2022 BWRX-300 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.3.2	Edited and added new information, including Table 2.7-1, Table 2.7-2, and Table 2.7-3, documented in the results and figures from the 2022 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.3.3	Deleted outdated information, made edits, and added new information, per in the results and figures in the 2023 FIA Report (Reference 2.7-38), and the 2022 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.4.1	Introduced the 2022 DNNP PSHA (Reference 2.7-41)
	Section 2.7.4.3	Added information and updated relevant figures based on information in the 2022 PSHA (Reference 2.7-41)
	Section 2.7.4.4	Added information and updated relevant figures based on the 2022 PSHA (Reference 2.7-41)
	Section 2.7.4.6	<ul style="list-style-type: none"> <li>This subsection is currently dedicated to present the results of the work performed in the 2022 PSHA report (Reference 2.7-41)</li> </ul> <p>Outdated information deleted</p>
	Section 2.7.4.7	<ul style="list-style-type: none"> <li>Added information under “Surface Faulting” based on findings reported in the 2022 geotechnical investigations (Reference 2.7-39)</li> <li>Added information on potential liquefaction based on the results reported in the 2022 Soil Liquefaction Assessment report (Reference</li> </ul>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
		2.7-42) and the 2022 PSHA (Reference 2.7-41) Added new figures
	Section 2.7.4.8	<ul style="list-style-type: none"> <li>This subsection is discontinued</li> </ul> <p>Previous information in Subsection 4.7.4.8 pf Revision 1 was merged into other Subsections of Section 2.7</p>
	Section 2.7.5	<ul style="list-style-type: none"> <li>Added referencing to the 2022 NK054-REP-01210-00175 Phase I Geotechnical Investigations (Reference 2.7-39) and the 2023 DNNP FIA report (Reference 2.7-38)</li> <li>Focus is on providing DNNP and BWRX-300 characteristics and parameters</li> </ul> <p>Information on “Bounding Design” is deleted as such information is detailed in Chapter 3, Section 3.3.1.1</p>
	Section 2.7.5.1	<p>New information added and updates made based on the 2023 FIA (Reference 2.7-38) and the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39); ); including:</p> <ul style="list-style-type: none"> <li>2.7.5.1.2 Bearing Capacity Evaluation for Proposed Foundations</li> <li>2.7.5.1.3 Earth Pressure</li> </ul> <p>2.7.5.1.4 Time-Dependent Deformation for Proposed Foundations</p>
	Section 2.7.5.2 and associated Subsections	<p>Outdated information deleted, new information added, and updates made based on the 2023 FIA (Reference 2.7-38) and the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39); including:</p> <ul style="list-style-type: none"> <li>2.7.5.2.1 Subgrade Profiles Stratigraphy</li> <li>2.7.5.2.2 Equivalent Linearized Static Properties of Soil and Engineered Fill Materials</li> <li>2.7.5.2.3 Equivalent Linearized Static Properties of Rock</li> <li>2.7.5.2.4 Dynamic Subgrade Properties</li> <li>2.7.5.2.5 Seismic Design Parameters</li> </ul> <p>2.7.5.2.6 Groundwater Level</p>
	Section 2.7.5.3	<p>New information added and updates made based on the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39)</p>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
	Previous Section 2.7.5.4	Information in Revision 1, Subsection 2.7.5.4 titled “Site Response Analysis” is deleted, since it is covered in Chapter 3, Subsection 3.3.1.1.2
	Previous Section 2.7.5.5	Information in Revision 1, Subsection 2.7.5.5 titled “Design Response Spectra for BWRX-300 at DNNP Site” is deleted and replaced with new information in Subsection 2.7.5.2.5.1 on Ground Motion Spectra
	Previous Section 2.7.5.6	Information in Revision 1, Subsection 2.7.5.6 on “Strain-Compatible Subgrade Profiles for BWRX-300 at DNNP Site” is deleted and replaced with new information in Subsection 2.7.5.2.5.2 on Strain-Compatible Soil properties
	Section 2.7.6	<ul style="list-style-type: none"> <li>Due to the reconfiguration of Section 2.7, several references in Revision 1 are deleted since they are not referenced anymore in Revision 2. The previous identifying numbers of such Revision 1 references were 2.7-20, -22, -23, -27, -28, -29, -33, -38, -39, -40, -42, -43, -44, -45, -46, -47</li> </ul> <p>New references added, from the current Reference 2.7-31 to Reference 2.7-43, inclusive</p>
	Section 2.8	Added Bullet number 6 for and edited the text based on the information in the 2022 DNNP EIS (Reference 2.8-10)
	Section 2.8.7	Added (Reference 2.8-10) regarding the 2022 DNNP EIS
	Section 2.9	Added a new bullet for and edited the text based on the information in the 2022 DNNP EIS (Reference 2.9-16)
	Section 2.9.3	Added (Reference 2.9-16) regarding the 2022 DNNP EIS
	Section 2.10	Introduced and added Table 2.10-1 titled Summary of DNNP Site Relevant Characteristics and Parameters
	Section 2.11.3	Introduced the work completed on FIA (Reference 2.11-19) and the Geotechnical investigations in (Reference 2.11-20)

### ACRONYM LIST

Acronym	Explanation
3D	Three-Dimensional
AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Earthquake
BL-AOO	Baseline Abnormal Operational Occurrence
BWR	Boiling Water Reactor
BWRX-300	Boiling Water Reactor, 10 <sup>th</sup> Design – 300 MWe
CANDU	CANada Deuterium Uranium
CAV	Cumulative Absolute Velocity
CB	Control Building
CEUS	Central Eastern United States
CGD	Canadian Geodetic Datum
CNEP	Consolidated Nuclear Response Plan
CNSC	Canadian Nuclear Safety Commission
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
DRL	Derived Release Limit
DSA	Deterministic Safety Analysis
DWMF	Darlington Waste Management Facility
EA	Environmental Assessment
EIS	Environmental Impact Statement
EME	Emergency Mitigating Equipment
EMP	Environmental Monitoring Program
EPRI	Electric Power Research Institute
ERA	Environmental Risk Assessment
FHA	Fire Hazards Assessment



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
FIA	Foundation Interface Analysis
FPC	Fuel Pool Cooling and Cleanup System
HCSC	Hazard-Consistent, Strain-Compatible
HU	Hydrostratigraphic Unit
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	ICS Pool Cooling and Cleanup System
ICS	Isolation Condenser System
INPO	Institute of Nuclear Power Operations
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LTC	Licence to Construct
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
NHS	Normal Heat Sink
NSCA	Nuclear Safety and Control Act
OPG	Ontario Power Generation
PCW	Plant Cooling Water System
PEOC	Provincial Emergency Operations Centre
PIE	Postulated Initiating Event
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PNERP	Provincial Nuclear Emergency Response Plan
PNGS	Pickering Nuclear Generating Station
POSAR	Pre-Operational Safety Analysis Report
PPE	Plant Parameter Envelope
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Assessment
RB	Reactor Building
RPV	Reactor Pressure Vessel
RWB	Radwaste Building

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
SA	Severe Accident
SAA	Severe Accident Analysis
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SCR	Secondary Control Room
SMR	Small Modular Reactor
SPT	Standard Penetration Test
SRA	Site Response Analysis
SSI	Soil-Structure Interaction
SSC	Structures, Systems, and Components
TB	Turbine Building
TLD	Thermoluminescent Dosimeter
UCS	Uniaxial Compression Stress
UHRs	Uniform Hazard Response Spectrum
USNRC	United States Nuclear Regulatory Commission
WPCP	Water Pollution Control Plant

## TABLE OF CONTENTS

<b>2.0</b>	<b>SITE CHARACTERISTICS .....</b>	<b>2-1</b>
2.1	Geography and Demography .....	2-5
2.1.1	Darlington Nuclear Site Context and Surrounding Land Uses .....	2-8
2.1.2	BWRX-300 Facility Layout and Exclusion Zone .....	2-10
2.1.3	Population Distribution and Density .....	2-12
2.1.4	Municipal Services .....	2-12
2.1.5	Site Access and Transportation Networks .....	2-13
2.1.6	Public Transit .....	2-13
2.1.7	Active Hiking and Cycling Trails.....	2-14
2.1.8	Park Spaces and Waterbodies .....	2-14
2.1.9	Industrial Facilities .....	2-14
2.1.10	References.....	2-14
2.2	Evaluation of Site-Specific Hazards .....	2-22
2.2.1	Introduction .....	2-22
2.2.2	External Hazards Evaluation Methodology .....	2-26
2.2.3	Characterization of Hazards from Transportation Accidents, Including Toxic Chemicals or Gas Releases/Explosions.....	2-27
2.2.4	Characterization of Stationary Non-Nuclear Accidents Hazards.....	2-31
2.2.5	Characterization of Stationary Nuclear Accidents Hazards .....	2-32
2.2.6	Characterization Industrial Hazards .....	2-35
2.2.7	Characterization of Biological, Animal, and Frazil Ice Hazards.....	2-35
2.2.8	Characterization of Ice Storm Hazard .....	2-36
2.2.9	Characterization of Electromagnetic Interference Hazard .....	2-37
2.2.10	Characterization of On-site Methane Hazard.....	2-37
2.2.11	References.....	2-38
2.3	Proximity of Industrial, Transportation and Other Facilities.....	2-39
2.3.1	Transportation Network.....	2-39
2.3.2	Industrial Facilities .....	2-39
2.3.3	Darlington Nuclear Generating Station Site .....	2-40
2.3.4	References.....	2-40
2.4	Plant Site Activities Influencing Plant Safety .....	2-41
2.4.1	Site Hazards .....	2-41
2.4.2	Measures of Site Protection.....	2-45

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.4.3	References.....	2-45
2.5	Hydrology .....	2-46
2.5.1	Introduction .....	2-46
2.5.2	Description of Heat Removal Methods and Heat Sink .....	2-50
2.5.3	Description of Potential Sources of Flooding .....	2-53
2.5.4	Potential Effects of Climate Change .....	2-64
2.5.5	Groundwater .....	2-65
2.5.6	Surface Water .....	2-66
2.5.7	References.....	2-68
2.6	Meteorology .....	2-70
2.6.1	Introduction .....	2-70
2.6.2	Temperature .....	2-74
2.6.3	Humidity .....	2-75
2.6.4	Rainfall / Precipitation .....	2-75
2.6.5	Wind and Wind Speed .....	2-76
2.6.6	Tornadoes and Hurricanes .....	2-79
2.6.7	Waterspouts .....	2-81
2.6.8	Dust Storms and Sandstorms .....	2-81
2.6.9	Snow and Ice Load, Freezing Rain, and Ice Storm .....	2-81
2.6.10	Lightning .....	2-82
2.6.11	Windborne Debris .....	2-83
2.6.12	Climate Change Impact .....	2-83
2.6.13	References.....	2-84
2.7	Geology, Seismology and Geotechnical Engineering .....	2-85
2.7.1	Site Location and Description .....	2-86
2.7.2	Geological Characteristics .....	2-86
2.7.3	Geotechnical Characteristics .....	2-89
2.7.4	Seismology Characteristics.....	2-98
2.7.5	Geotechnical and Seismological Parameters .....	2-110
2.7.6	References.....	2-124
2.8	Site Characteristics Impact on Dispersion of Radioactive Material.....	2-178
2.8.1	Impact of Meteorology and Climate .....	2-182
2.8.2	Impact of Hydrology and Hydrogeology.....	2-186
2.8.3	Impact of Land Cover and Use .....	2-188

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.8.4	Impact of Population .....	2-190
2.8.5	Impact of Accident Scenarios and Dispersion Models .....	2-193
2.8.6	Impact of Biological Data .....	2-193
2.8.7	References.....	2-193
2.9	Radiological Conditions Due to External Sources.....	2-194
2.9.1	Radiological Conditions in the Environment .....	2-196
2.9.2	Radiation Monitoring Systems .....	2-200
2.9.3	References.....	2-205
2.10	Site-Related Issues in Emergency Preparedness and Response and Accident Management .....	2-206
2.10.1	General Consideration .....	2-207
2.10.2	Feasibility of Emergency Preparedness and Response .....	2-207
2.10.3	Evacuation Time Estimates and Route .....	2-208
2.10.4	Support Networks in the Vicinity of the Site .....	2-209
2.10.5	Administrative Measures with External Organizations .....	2-211
2.10.6	References.....	2-214
2.11	Monitoring of Site-Related Parameters .....	2-215
2.11.1	Volcanic Phenomena Monitoring .....	2-216
2.11.2	Surface Faulting Monitoring .....	2-216
2.11.3	Seismic and Geotechnical Monitoring.....	2-216
2.11.4	Meteorological Monitoring.....	2-218
2.11.5	Hydrological Monitoring .....	2-218
2.11.6	Radiation Monitoring .....	2-219
2.11.7	Environmental Monitoring .....	2-219
2.11.8	Biological Organisms and Human Induced Hazards Monitoring.....	2-219
2.11.9	Long Term Monitoring Program .....	2-220
2.11.10	References.....	2-221
2.12	Ongoing Work Plans .....	2-223
2.12.1	Introduction .....	2-223
2.12.2	Foundation Interface Analysis.....	2-229
2.12.3	Site Geotechnical and Seismic Hazard Investigation Plan .....	2-231
2.12.4	Flood Hazard Assessment.....	2-233
2.12.5	Climate Change Impact .....	2-235
2.12.6	3-Second Wind Gust Speed .....	2-237
2.12.7	Snow Load and Coincident Winter Probable Maximum Precipitation ...	2-238

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.12.8	Confirmation of Probable Maximum Precipitation .....	2-239
2.12.9	References.....	2-240
2.13	Appendices .....	2-241

## LIST OF TABLES

Table 2.1-1: Site Layout, Geographic, and Demographic Characteristics and Parameters .....	2-6
Table 2.1-2: Population Data for the Municipality of Clarington for 2021 .....	2-12
Table 2.2-1: Screening and Validation of CNSC–Identified DNNP Site-Specific Hazards.....	2-23
Table 2.4-1: Summary of Gases and Chemicals Stored on DNNP Site .....	2-42
Table 2.5-1: Hydrological Characteristics Summary of DNNP Site and Surrounding Area ....	2-47
Table 2.5-2 Key Modelling and Assessment Parameters for Darlington Creek and On-site External Flood Hazards (Reference 2.5-18) .....	2-61
Table 2.5-3 Summary of Lake Ontario Depth Averaged current speed and direction (Reference 2.5-9).....	2-67
Table 2.6-2: DNGS Design Basis Durations at Low Temperature Applicable to DNNP .....	2-74
Table 2.6-3: Extreme 3-second Gust Speeds for the DNNP (Reference 2.6-14) .....	2-78
Table 2.6-4: Extreme 3-second Gust Speeds Envelop Based on Four-station Data Around the DNNP (Reference 2.6-14) .....	2-78
Table 2.6-5: DNNP Site Characteristics for Design Basis Tornado (Reference 2.6-9).....	2-80
Table 2.6-6: DNNP Site Tornado Missiles Spectrum for Maximum Horizontal Speed (Reference 2.6-9).....	2-80
Table 2.6-7: Snow Loads and Winter PMP Values for DNGS and DNNP (Reference 2.6-15).....	2-82
Table 2.7-1: Stratigraphic Units for the DNNP Site	2-92
Table 2.7-2: In-situ Soil Units Stratigraphy under the Power Block (Reference 2.7-39) .....	2-93
Table 2.7-3: Rock Units Stratigraphy (Reference 2.7-39).....	2-94
Table 2.7-4: Figures Presenting UHRS Based on Mean Hazard Results.....	2-106
Table 2.7-5 Deformation for Proposed Foundations (Reference 2.7-38).....	2-113
Table 2.7-6 Consolidation Settlement Method (Reference 2.7-38).....	2-114
Table 2.7-7: As-Built Linearized Static Properties for Soil Layers (Reference 2.7-39) .....	2-117
Table 2.7-8: Summary of Linearized Static Rock Properties (Reference 2.7-39) .....	2-118
Table 2.7-9a: Rock Dynamic Properties (Reference 2.7-39) .....	2-119
Table 2.7-9b: Rock Dynamic Properties (Reference 2.7-39) .....	2-119
Table 2.7-10a: In-situ Soil Dynamic Properties (Reference 2.7-39) .....	2-120
Table 2.7-10b: In-situ Soil Dynamic Properties (Reference 2.7-39) .....	2-121
Table 2.7-11 Samples of Groundwater Elevation and vertical Hydraulic Gradients, BH12 Area (Reference 2.7-39, Table 3-7).....	2-123
Table 2.8-2: Temperature Normals Near the Darlington Nuclear Site (Reference 2.8-2).....	2-183
Table 2.8-3: Precipitation at Bowmanville Mostert Station (1981-2010) .....	2-184

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Table 2.8-4: Darlington Nuclear – 2021 Annual Average Wind Frequency by Direction (at 10 m height) .....	2-185
Table 2.8-5: Population Distribution Surrounding Darlington Nuclear Site Based on 2016 Census Data (Reference 2.8-2).....	2-192
APPENDIX A – List of Industrial Facilities within the Survey Area .....	2-242
APPENDIX B – List of Roads within the Survey Area .....	2-244
APPENDIX C – List of Park Spaces and Water Bodies within the Survey Area.....	2-265



## LIST OF FIGURES

Figure 2.1.1-1: Darlington Nuclear Site Proximity to Pickering Nuclear Generating Station...	2-16
Figure 2.1.1-2: Darlington Nuclear Generation Station and Darlington New Nuclear Project Lands .....	2-17
Figure 2.1.1-3: Darling Nuclear Generating Station and – Darlington New Nuclear Project Proximity to Industry .....	2-18
Figure 2.1.1-4: DNNP Regional Study Area .....	2-19
Figure 2.1.3-1: Clarington Urban Areas .....	2-20
Figure 2.1.8-1: Darlington Nuclear Site – Active Darlington Waterfront Trail.....	2-21
Figure 2.5.3.2-1 Pre-development Darlington Nuclear Site Drainage (Reference 2.5-18).....	2-59
Figure 2.5.3.2-2 Post-development Darlington Nuclear Site Drainage (Reference 2.5-18)....	2-60
Figure 2.7.1-1: Existing Topographic Contours of DNNP Site (Reference 2.7-26).....	2-127
Figure 2.7.2-1: Darlington Nuclear Site - Regional Surficial Geology (Reference 2.7-1).....	2-128
Figure 2.7.2-2: Darlington Nuclear Site - Regional Bedrock Geology (Reference 2.7-1).....	2-129
Figure 2.7.2-3: Offshore Geophysics Bathymetry Hillshade Map (Reference 2.7-40).....	2-130
Figure 2.7.2-4: Offshore Geophysical Bathymetric Contour Map (Reference 2.7-40) .....	2-131
Figure 2.7.2-5: Lakebed Bathymetric Contours along DNNP Site's Shoreline (Reference 2.7-2) .....	2-132
Figure 2.7.2-6: Lakebed Surface Geology Map along DNNP Site's Shoreline (Reference 2.7-2) .....	2-133
Figure 2.7.3.1-1: Locations of CH2MHILL (2009) Monitoring Wells/Boreholes (Reference 2.7-1) .....	2-134
Figure 2.7.3.1-2: Locations of AMEC (2012) Boreholes (Reference 2.7-35) .....	2-135
Figure 2.7.3.1-3: Locations of EXP (2013) Boreholes (Reference 2.7-36) .....	2-136
Figure 2.7.3.1-4: Stratigraphic Profile near BWRX-300 Location (Reference 2.7-36) .....	2-137
Figure 2.7.3.1-5: Stratigraphic Profile near Topographic Ridge East of the BWRX-300 Location (Reference 2.7-36).....	2-138
Figure 2.7.3.1-6: Boreholes Sections (1-1' and 2-2') Locations Plan (Phase 1) (Reference 2.7-39) .....	2-139
Figure 2.7.3.2-1: Subsurface Stratigraphic Profile at Cross-Section 1-1 (Reference 2.7-39)	2-140
Figure 2.7.3.2-2: Subsurface Stratigraphic Profile at Cross-Section 2-2 (Reference 2.7-39)	2-141
Figure 2.7.3.2-3: Unit 3 Groundwater Levels – Shallow/Water Table (Reference 2.7-39).....	2-142
Figure 2.7.3.2-4: Unit 4a Groundwater Flow – Interglacial Deposits (Reference 2.7-39) .....	2-143
Figure 2.7.3.2-5: Unit 5 Groundwater Flow – Shallow Bedrock (Reference 2.7-39).....	2-144
Figure 2.7.3.2-6: Units 6a – 6b Groundwater Flow – Shallow Bedrock (Reference 2.7-39) ..	2-145
Figure 2.7.3.2-7: Regional Groundwater Levels – Shallow/Water Table (Reference 2.7-1) ..	2-146

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 2.7.3.2-8: Regional Groundwater Flow – Interglacial Deposits (Reference 2.7-1).....	2-147
Figure 2.7.3.2-9: Regional Groundwater Flow – Shallow Bedrock (Reference 2.7-1) .....	2-148
Figure 2.7.4.2-1: Principal Subdivisions of Precambrian Rocks in the Great Lakes Region (Reference 2.7-5) .....	2-149
Figure 2.7.4.2-2: Regional Source Zones for IRM/SGR Boundary Eastern Boundary (Reference 2.7-41) .....	2-150
Figure 2.7.4.3-1: Map of Independent Earthquakes in the Updated Earthquake Catalogue for the Study Region (Reference 2.7-41).....	2-151
Figure 2.7.4.4-1: Logic Tree for Distributed Seismicity Sources (Reference 2.7- 42).....	2-152
Figure 2.7.4.4-2: Regional Seismicity Source Zones from the Geological Survey of Canada's Sixth Generation H Model (Plate 13 in Reference 2.7-41) .....	2-153
Figure 2.7.4.4-3: Regional Seismicity for Zoneless Model (Plate 14 in Reference 2.7-41)....	2-154
Figure 2.7.4.4-4: Local Source Zones (Plate 15 in Reference 2.7-41) .....	2-155
Figure 2.7.4.6-1: Horizontal UHRS at Elevation 52.93 m Based on Mean Hazard (Reference 2.7-41) .....	2-156
Figure 2.7.4.6-2: Vertical UHRS at Elevation 52.93 m Based on Mean Hazard (Reference 2.7-41) .....	2-157
Figure 2.7.4.6-3: Horizontal UHRS at Elevation 64 m Based on Mean Hazard (Reference 2.7-41) .....	2-158
Figure 2.7.4.6-4: Vertical UHRS at Elevation 64 m Based on Mean Hazard (Reference 2.7-41) .....	2-159
Figure 2.7.4.6-5: Horizontal UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41).....	2-160
Figure 2.7.4.6-6: Vertical UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41).....	2-161
Figure 2.7.4.6-7: Horizontal UHRS for Reference Rock Based on Mean Hazard (Reference 2.7-41) .....	2-162
Figure 2.7.4.6-8: Vertical UHRS for Reference Rock Basedon Mean Hazard (Reference 2.7-41) .....	2-163
Figure 2.7.4.6-9: Initial Horizontal DBE and BDBE Spectra for Reactor Base (Elevation 52.93 m) (Reference 2.7-41) .....	2-164
Figure 2.7.4.6-10: Initial Horizontal DBE and BDBE Spectra for Soil-Rock Interface (Elevation 64 m) (Reference 2.7-41) .....	2-165
Figure 2.7.4.6-11: Initial Horizontal DBE and BDBE Spectra for Finished Grade (Elevation 88 m) (Reference 2.7-41) .....	2-166
Figure 2.7.4.6-12: Comparison of Finished Grade Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41) .....	2-167
Figure 2.7.4.6-13: Comparison of Reactor Building Base Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41).....	2-168

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 2.7.4.7-1: Locations of Boreholes Potentially to Liquefaction Under the DBE Event (Reference 2.7-42) .....	2-169
Figure 2.7.4.7-2: Locations of Boreholes Potentially to Liquefaction Under the BDBE Event (Reference 2.7-42) .....	2-170
Figure 2.7.5.1.3-1: Rock and Soil Pressures Around RB Wall (Reference 2.7-38).....	2-171
Figure 2.7.5.2.5-1: DBE and BDBE Foundation Input Response Spectra for Reactor Base (Elevation 52.93 m) (Reference 2.7-41) .....	2-172
Figure 2.7.5.2.5-2: DBE and BDBE Performance Based Intermediate Response Spectra for Soil/Rock Interface (Elevation 64 m) (Reference 2.7-41).....	2-173
Figure 2.7.5.2.5-3: DBE and BDBE Performance Based Surface Response Spectra for Finished Grade (Elevation 88 m) (Reference 2.7-41).....	2-174
Figure 2.7.5.2.5-4: Subgrade Profiles of DBE and BDBE HCSC Shear Wave Velocities (Reference 2.7-41) .....	2-175
Figure 2.7.5.2.5-5: Subgrade Profiles of DBE and BDBE HCSC Compression Wave Velocities (Reference 2.7-41) .....	2-176
Figure 2.7.5.2.5-6: Subgrade Profiles of DBE and BDBE HCSC Damping Ratios (Reference 2.7-41) .....	2-177
Figure 2.9-1: Darlington Nuclear Site Critical Groups and Environmental Monitoring Locations (Reference 2.9-2) .....	2-204
Figure 2.10.2-1: Darlington Nuclear Site Showing DNGS and DNNP Areas .....	2-212
Figure 2.10.2-2: Area of Consideration for Traffic Management Plan .....	2-213
Figure 2.12.2-1: FIA Modelling Workflow and Deliverables .....	2-230
Figure 2.12.3-1: Workflow for the Geotechnical Program.....	2-233
Figure 2.12.5-1: Risk Roadmap for OPG Strategy on Addressing Climate Change Impacts	2-236

## 2.0 SITE CHARACTERISTICS

Information in Chapter 2 details the site characteristics and their evaluation in support for the design, safety assessment and periodic safety review (Reference 2.0-4) of the Boiling Water Reactor, 10<sup>th</sup> Design – 300 MWe (BWRX-300) facility (also known as BWRX-300 facility). Over the planned design life (refer to Chapter 1, Table 1.5-1) of the BWRX-300 facility, the information in Chapter 2 will periodically be updated (Reference 2.0-4) to risk-inform the evaluation and implications of any such updates on safety.

Chapter 2 includes the following characteristics of Ontario Power Generation's (OPG) Darlington New Nuclear Project (DNNP) site and the surrounding region:

- Geography and Demography (Section 2.1)
- Evaluation of Site-specific Hazards (Section 2.2)
- Proximity of Industrial, Transportation and Other Facilities (Section 2.3)
- Plant Site Activities Influencing Plant Safety (Section 2.4)
- Hydrology (Section 2.5)
- Meteorology (Section 2.6)
- Geology, Seismology, and Geotechnical Engineering (Section 2.7)
- Potential Effects of Nuclear Power Plants in the Region (Section 2.8)
- Radiological Conditions due to External Sources (Section 2.9)
- Site-related Issues in Emergency Preparedness and Response, and Accident Management (Section 2.10)
- Monitoring of Site-related Parameters (Section 2.11)

Chapter 2 also includes Section 2.12 which describes OPG's disposition plans to finalize remaining DNNP site-specific characterization work including, for example, Foundation Interface Analysis (FIA), confirmatory site geological and seismic hazard investigations, and climate change effects on-site hydrological and meteorological parameters.

The following key chapters should be referred for additional information relevant to the material reported in Chapter 2:

1. Chapter 1: Introduction and General Considerations

Information in Chapter 1, Sections 1.4 and 1.5 describes the DNNP site layout, as well as the BWRX-300 facility footprint, key parameters, and basic dimensions of key buildings in the Power Block.

2. Chapter 3: Safety Objectives and Design Rules for Structures, Systems, and Components

Chapter 3, Section 3.3 includes information on the BWRX-300 design approach to prevent and mitigate the effect of external hazard on safety-classified structures, systems, and components (SSCs). Also, Chapter 3, Subsection 3.5.5.2 describes the design loads and load combinations on the deeply embedded Reactor Building (RB) structure.

3. Chapter 6: Engineered Safety Features

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Information is provided in Chapter 6, Section 6.2 on the design of the Isolation Condenser System; and in Section 6.4 on the BWRX-300 control room habitability features including missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, and fire protection.

4. Chapter 7: Instrumentation and Control

Measures for fire protection and qualification for electromagnetic compatibility are described in Chapter 7.

5. Chapter 9A: Auxiliary Systems

Chapter 9A presents information on the BWR-X-300 fuel storage and handling system in Subsection 9A1.2, Fuel Pool Cooling and Cleanup System (FPC) in Subsection 9A1.3, Plant Cooling Water System (PCW) in Subsection 9A.2.1, Normal Heat Sink (NHS) in Subsection 9A.2.5, Isolation Condenser System Pool Cooling and Cleanup System (ICC) in Subsection 9A.2.6, Heating, Ventilation, and Air Conditioning (HVAC) Systems in Section 9A.5, Fire Protection Systems, in Section 9A.6.

6. Chapter 9B: Civil Engineering Works and Structures

General design requirement information is provided in Chapter 9B, Section 9B.2 on the integrated RB, and Section 9B.3 on other structures including other buildings in the Power Block, the Pumphouse/Forebay as well as the intake and discharge tunnels.

7. Chapter 10: Steam and Power Conversion Systems

In Chapter 10, information related to equipment functions, design basis, operation, and maintenance is presented in Section 10.5 for the Main Condenser and Auxiliaries (MCA) system, and in Section 10.8 for the Circulating Water System (CWS).

8. Chapter 15: Safety Analysis

Chapter 15, Subsection 15.5.3 documents the Deterministic Safety Analysis (DSA) of bounding Baseline Abnormal Operational Occurrences (BL-AOOs), while 15.5.4 evaluates the bounding BWRX-300 Design Basis Accidents (DBAs) involving Loss-of-Coolant Accidents (LOCA) and non-LOCA. Also, Subsections 15.5.5 and 15.5.6 present analyses of Design Extension Conditions (DECs) with and without core damage, respectively. Furthermore, Subsection 15.6.1 described the general approach to the Probabilistic Safety Analysis (PSA) while Section 15.7 includes results of analyzed DSA and PSA bounding events. Finally, Appendix 15A demonstrates implementing Defence-in-Depth (D-in-D) provisions ensures protection against unacceptable radiation releases

9. Chapter 19: Emergency Preparedness and Response

The development of the DNNP nuclear emergency response plan is presented in Section 19.1, the emergency response facilities are described in Section 19.2, and the accident assessment techniques are detailed in Section 19.3.

10. Chapter 20: Environmental Aspects

Chapter 20 describes OPG's Environmental Monitoring Program in Subsection 20.11.2, Effluent Monitoring Program in Subsection 20.11.3, and Groundwater Monitoring Program in Subsection 20.11.4.

11. BWRX-300 Security Annex

The prescribed information in the Security Annex documents the analysis of a large commercial aircraft crash.

## Scope

Chapter 2 scope includes the establishment of site characteristics that comprise information such as:

1. The site location, the area under control of OPG, and the area surrounding the DNNP site including activities which impact BWRX-300 facility operation, population distribution and density (Section 2.1), and the locations and transport routes that present potential risk for the facility (Section 2.3).
2. The site-specific external hazard evaluation (Section 2.2) for events of natural and human-induced origin during the planned lifetime of the facility, and any process or activity at the site that affects the operation of the facility (Section 2.4).
3. The collection of DNNP site-specific baseline data such as hydrological (Section 2.5); meteorological (Section 2.6); as well as geological, seismological, geotechnical (Section 2.7) information.
4. The description of the site and the surrounding environment (Sections 2.8), and of external sources related to the dispersion of radioactive material in air, water, and soil (Section 2.9).
5. The feasibility of emergency preparedness as related to accessibility and transport of any pertinent equipment to the DNNP site and the BWRX-300 facility (Section 2.10).
6. The arrangements for monitoring site-related parameters (Section 2.11) throughout the lifetime of the facility.

## Relevant Legislations and Regulations

The following provisions of the Nuclear Safety and Control Act (Reference 2.0-1), the General Nuclear Safety and Control Regulations (Reference 2.0-2) and the Class I Nuclear Facilities Regulations (Reference 2.0-3) are relevant to Chapter 2.

- Subsection 44(1) of the NSCA (Reference 2.0-1) states that “[t]he Commission may, with approval of the Governor in Council, make regulations.  
(e) Respecting the location, design, construction, installation, operation, maintenance, modification, decommissioning, abandonment and disposal of a nuclear facility or part of a nuclear facility.  
(o) Establishing requirements to be complied with by any person who possesses, uses, packages, transports, stores, or disposes of a nuclear substance or prescribed equipment or who locates, designs, constructs, installs, operates, maintains, modifies, decommissions, or abandons a nuclear facility or nuclear-powered vehicle.
- Section 3 of the Class I Nuclear Facilities Regulations (Reference 2.0-3) states that “[a]n application for a licence in respect of a Class I nuclear facility, other than a licence to abandon, shall contain the following information in addition to the information required by Section 3 of the General Nuclear Safety and Control Regulations (Reference 2.0-2):
  - a. A description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone
  - b. Plans showing the location, perimeter, areas, structures, and systems of the nuclear facility
  - c. Proposed management system for the activity to be licensed, including measures to promote and support safety culture

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- d. Name, form, characteristics, and quantity of any hazardous substances that may be on the site while the activity to be licensed is carried on
- e. Proposed worker health and safety policies and procedures
- f. Proposed environmental protection policies and procedures
- g. Proposed effluent and environmental monitoring programs
- Section 5 of the Class I Nuclear Facilities Regulations (Reference 2.0-3) states that: “[a]n application for a licence to construct a Class I nuclear facility shall contain the following information in addition to the information required by Section 3:
  - a. Description of the proposed design of the nuclear facility, including the manner in which the physical and environmental characteristics of the site are considered in the design
  - b. Description of the environmental baseline characteristics of the site and the surrounding area
  - c. Effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects
  - d. Proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical, and radiological characteristics

**References**

- 2.0-1 Government of Canada, “Nuclear Safety and Control Act (S.C. 1997, c. 9).”
- 2.0-2 Government of Canada SOR/2000-202, “General Nuclear Safety and Control Regulations.”
- 2.0-3 Government of Canada SOR/2000-204, “Class I Nuclear Facilities Regulations.”
- 2.0-4 CNSC Regulatory Document REDGOC-2.3.3, “Operating Performance - Periodic Safety Reviews.”

## **2.1 Geography and Demography**

Section 2.1 details the geographical and demographical baseline characteristics of the DNNP site and the surrounding regions. It contains the following information:

- Darlington Nuclear site context and surrounding land uses - Subsection 2.1.1
- BWRX-300 facility layout and the exclusion zone - Subsection 2.1.2
- Population distribution and density - Subsection 2.1.3
- Municipal services - Subsection 2.1.4
- Site access and transportation networks - Subsection 2.1.5
- Public transit – Subsection 2.1.6
- Active hiking and cycling trails - Subsection 2.1.7
- Parks spaces and waterbodies - Subsection 2.1.8
- Industrial facilities - Subsection 2.1.9

Table 2.1-1 lists key geographic and demographic characteristics and parameters within a 10-km survey area surrounding the Darlington Nuclear site.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.1-1: Site Layout, Geographic, and Demographic Characteristics and Parameters**

Characteristic	Value/Description	
Land Size	Darlington Nuclear site	Approximately 4.9 km <sup>2</sup>
	DNNP	Approximately 1.8 km <sup>2</sup>
	DNGS	Approximately 3.1 km <sup>2</sup>
Exclusion Zone	BWRX-300	350 m (radius) from the RB outside wall
	DNGS	914 m
Topography	<ul style="list-style-type: none"> <li>Current parking and storage areas east of the DWMF is at approximately 88 m (Canadian Geodetic Datum of 1928 (CGVD28), or simply CGD))</li> <li>Further east, the terrain rises to 102 m CGD close to the Darlington Creek watershed</li> <li>Extreme berm of elevation from 100 to 110 characterize the north boundary of the southern portion of the site to the railway tracks The northern portion of the site is bounded the north by Energy Road and to the south by the Railway tracks</li> <li>East of Holt Road, the terrain peaks at 120 m CGD and slopes down to the east to roughly 86 m CGD</li> </ul>	
Grade Elevation	Plant (BWRX-300 Facility)	88 m CGD (Refer to Subsection 2.7.1)
Population Distribution and Density (2021), for the Municipality of Clarington	Courtice	28,545
	Bowmanville	47,176
	Orono	2,476
	Newcastle	11,933
	Total	90,130
Municipal Service within the 10-km Survey area	Fire Emergency Stations	6 (Excluding DNGS site fire station)
	Regional Police Station	One (plus one administrative police department)
	Hospitals	One (Lakeridge Health in Bowmanville)
Directly Adjacent Industrial Facilities	East	St. Marys Cement Group
	West	<ul style="list-style-type: none"> <li>Darlington Nuclear Energy Complex</li> <li>CoPart, Vehicle Auction Facility</li> <li>Covanta Durham York Energy Centre</li> <li>Courtice Water Pollution Control Plant (WPCP)</li> <li>East Penn, Batteries warehouse facility</li> <li>Future Anaerobic Digester facility</li> </ul>
Transportation network within 10 km	Highways	401, 407, 418
	Railways lines	<ul style="list-style-type: none"> <li>Canadian National, south of Highway 401 and bisects the site</li> <li>Canadian Pacific, north of Highway 401</li> </ul>
	Airports	Oshawa Executive Airport
	Naval Ports	Port of Oshawa East Pier

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Characteristic	Value/Description	
	Private Dock	Private dock on St. Marys facility
Public Transit	Bus (902A King bus line)	One stop at Old Holt Road and King Street
	Transit-on-demand	Request pick up to nearest transit stop
	Rural-on-demand	Request pick up at current location
	88 GO Bus	Multiple stops along Bowmanville Avenue and King Street
	GO Transit's Lakeshore East Rail Service (planned for operation in 2026)	Courtice GO Station Bowmanville GO Station
Hiking and Cycling Trails	Darlington Waterfront Trail	Pedestrian and cyclists trail
Parks Spaces and Waterbodies (Note: A complete list is provided in Appendix C)	Provincial Parks	One – Darlington Provincial Park
	Recreational Facilities	Darlington Hydro Soccer Field and Bowmanville Baseball Fields
	Conservation Areas	Five in Bowmanville and two in Oshawa
	Beaches	Three – Two in Bowmanville and one in Oshawa
Industrial Facilities within 10 km	<ul style="list-style-type: none"> <li>• Directly adjacent industrial facilities, refer to Subsection 2.1.1</li> <li>• A complete list of industrial facilities falling within the surveyed area is found in Appendix A.</li> <li>• Pickering Nuclear Generating Station, about 25 km west of DNNP</li> </ul>	

## **2.1.1 Darlington Nuclear Site Context and Surrounding Land Uses**

### **Site Topography**

The Darlington Nuclear site topography is briefly described in Subsection 2.7.1. The 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.1-9) provides in this Subsection 2.1.1 additional information on the site topography including key detailed terrain elevations, as briefly recapped in the following paragraph.

The Darlington Nuclear site is situated in an undulating to moderately rolling limestone till plain, although its natural contours have been extensively graded. The existing 4-unit Darlington Nuclear Generating Station (DNGS) is located at elevation of about 78 m CGD. This is the lowest elevation area of the southern portion of the Darlington Nuclear site. From this location, the site slopes upward to the northwest, north and east. To the east, the terrain steadily slopes upward along the Lake Ontario shoreline, forming a bluff. The DNNP site, currently a parking and storage area southeast of the Darlington Waste Management Facility (DWMF), is just north of shoreline bluff, at approximately 88 m CGD. Farther east, the terrain rises to elevation 102 m CGD at the boundary of the Darlington Creek watershed before sloping down to its main branch near the eastern boundary of the site. The north boundary of the southern portion of the Darlington Nuclear site is characterized by an extensive berm that ranges in elevation from 100 m CGD to 110 m CGD and separates the southern portion of the site from the transecting Canadian National Railway tracks. The northern portion of the site is bounded to the north by Energy Drive and to the south by the Canadian National Railway tracks. Between Crago Road and Park Road, there is a large ridge rising to 132 m CGD. Between Park Road and Holt Road, the terrain ranges from 98 m to 130 m CGD. East of Holt Road, the DNNP terrain peaks at 120 m CGD and slopes downward to the east to roughly 86 m CGD.

### **Area and Bounding Roads**

The Darlington Nuclear site is approximately 4.9 km<sup>2</sup> in size and located within the Municipality of Clarington, Regional Municipality of Durham, Province of Ontario, Canada. OPG also owns and operates the eight-unit Pickering Nuclear Generating Station (PNGS) (refer to Subsection 2.2.5.2) within the City of Pickering which is located approximately 25 km to the west of the Darlington Nuclear site, as shown in Figure 2.1.1-1.

The Darlington Nuclear site encompasses both the DNGS and the DNNP lands as shown in Figure 2.1.1-2. The Darlington Nuclear site is bounded by Crago Road to the west, Energy Drive to the north, St. Marys Cement to the east and Lake Ontario to the south. The existing DNGS site is approximately 3.1 km<sup>2</sup> in size and is located west of Holt Road on the western portion of the Darlington Nuclear site, whereas the DNNP land of approximately 1.8 km<sup>2</sup> is located east of Holt Road. Figure 2.1.1-2 shows also the 914-meter DNGS exclusion zone, which partly overlaps the location where the BWRX-300 first unit is to be built in the southwestern corner of the DNNP site as shown in Chapter 1, Figure A1.1-2.

### **Industrial Facilities**

The major industrial facilities in the vicinity of the Darlington Nuclear site, as shown in Figure 2.1.1-3, include:

1. St. Marys Cement Group which is located directly east of the DNNP site on Bowmanville Avenue, and is an active quarry for resources servicing the aggregate and concrete industry
2. The lands designated as Clarington Energy Business Park which is located directly west of the DNGS and includes:

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- a. Covanta Durham York Energy Centre which manages household waste from the regions of Durham and York
  - b. OPG's Darlington Energy Complex, an approximately 27,900 m<sup>2</sup> multi-use building that provides offices and services supporting the Darlington Refurbishment project
  - c. CoPart, a vehicle auction and recycling facility
  - d. East Penn, a warehousing facility for batteries
  - e. Courtice Water Pollution Control Plant (WPCP), a wastewater treatment facility commissioned in late 2007, with an average day rated capacity of 68.2 million liters per day with a peak flow capacity of 180 million liters per day (Reference 2.1-7)
  - f. Planned location for a project that is being evaluated involving an Anaerobic Digester facility (Reference 2.1-7) to treat raw sludge collected from Courtice WPCP
3. OWASCO RV, which is a recreational vehicle sale and service centre, located north of Highway 401

There are some industrial developments in the Courtice Employment Area located northwest of the Darlington Nuclear site, including warehousing and automobile dealerships. All of the industrial facilities falling within the surveyed area are listed in Appendix A.

### **Developmental Activities**

OPG actively reviews planning applications in the Municipality of Clarington to monitor sensitive land use developments within 3 km of the DNGS and DNNP facilities. Additionally, OPG reviews planning applications within 10 km of the Darlington Nuclear site in the Municipality of Clarington and the City of Oshawa. These applications include official plan amendments, zoning by-law amendments, draft plans of subdivision and condominium, and other miscellaneous planning related documents.

OPG completes an annual development activity report detailing all proposed developments in the municipalities of Clarington and Oshawa within 10 km of the Darlington Nuclear site. In such a report, OPG reviews the:

- a. Type and location of proposed application
- b. Date on which the application was submitted
- c. Details of the proposed application
- d. Status of the application

### **Urban Communities and Rural Areas**

The urban communities of Oshawa and Courtice are located northwest of the Darlington Nuclear site, while the urban community of Bowmanville is located to the northeast of the DNNP site. A rural area separating the Clarington urban areas of Courtice and Bowmanville is located immediately north of the DNNP site. The community of Newcastle is also located east of the DNNP site within the survey area; albeit only a portion is included in the survey area. For the purposes of Section 2.1 and Section 2.3, the geographic limits defined for the survey area are approximately 10 km from the site and include Taunton Road to the north, Simcoe Street to the west, an approximate border of Darlington Clarke Townline Road to the east, and Lake Ontario to the south (refer to Figure 2.1.1-4).

## **Land Use Assessment for Environmental Effects**

The 10 km survey area is consistent with the Land Use Assessment Zone, which was the furthest distance that measurable effects on planned land use structure as well as impacts on sensitive land uses are identified in the proximity to the Darlington Nuclear site. The Land Use Assessment of Environmental Effects Technical Support Document completed in 2009 identified the Regional Study Area as being approximately 50 km from the Darlington Nuclear site as shown in Figure 2.1.1-4. The DNNP Land Use Environmental Assessment Follow-Up Monitoring Plan / Methodology Report was developed in 2022 NK054-CORR-00531-10635 (Reference 2.1-3) to fulfill the requirement of OPG Commitment D-P-12.7 in the 2021 NK054-REP-01210-00078 (Reference 2.1-2). As per the 2022 NK054-CORR-00531-10635 (Reference 2.1-3), OPG will continue to monitor planning development in land use in proximity to the DNNP site, and regularly consult with the Municipality of Clarington, City of Oshawa and the Regional Municipality of Durham on proposed land use changes. The effects on implementation of emergency plans will be investigated throughout the site preparation and construction phases.

### **2.1.2 BWRX-300 Facility Layout and Exclusion Zone**

The layouts of the DNNP site and BWRX-300 Unit 1 as well as associated infrastructures are described in Chapter 1, Section 1.4, and Section 1.5 satisfy the regulatory requirements of Sections 4.5.4 and 4.5.5 of REGDOC-1.1.2 (Reference 2.1-1). The selected location, in the southwestern corner of the DNNP area, limits the amount of spoilage to remove and avoids encroachment on the Bank Swallow habitat. This location is also in proximity to DNGS ensuring effective connections to DNGS available infrastructure. The DNNP site also incorporates considerations that support a total of four BWRX-300 units, as conceptually shown in Figure 5 of the 2022 DNNP BWRX-300 Environmental Impact Statement (EIS) NK054-REP-07730-00055 (Reference 2.1-4).

The deployment of the BWRX-300 facility does not require expanding the DNGS switchyard. Rather, a new 230 kV switchyard is to be located East of the Extended Holt Rd and South of the Canadian National Railway tracks, adjacent to the BWRX-300 facility buildings, as shown in Chapter 1, Figure A1.1-2 and Figure A1.4-2 for one unit and conceptually shown in Figure 7 of the 2022 NK054-REP-07730-00055 (Reference 2.1-4) for four units.

Existing roads are being used to the maximum extent practicable and no new off-site roadways are required.

The Pumphouse/Forebay structure is positioned outside the northwestern corner of the protected area. As described in Chapter 9B, Subsection 9B.3.5.2, onshore vertical shafts are designed to facilitate the operation of up to four BWRX-300 units and the construction of the intake and discharge tunnels. The intake tunnel conveys cooling water from the lakebed intake structure to the onshore intake vertical shaft. The discharge tunnel conveys the discharge water from the onshore discharge vertical shaft to the discharge tunnel and diffusers. The discharge structure is located near the lakeshore and does not require lake infill.

#### **2.1.2.1 Required Exclusion Zones**

The exclusion zone is established at 350 m from the RB outside wall. For the BWRX-300 first unit, the exclusion zone partly overlaps the eastern portion of the DNGS site, as shown in Chapter 1, Figure A1.1-2. The exclusion zone of 350 m for the conceptual layout of four units shown in Figure 5 of the 2022 NK054-REP-07730-00055 (Reference 2.1-4) is within the DNNP eastern boundary with St. Mays Cement industrial facility.

The rationale for determining the exclusion zone is discussed in Section 8 of the 2022 NK054-REP-01210-00142 (Reference 2.1-5), and considers the security requirements, evacuation

needs, land usage needs, and environmental conditions, in accordance with Section 6.5 of REGDOC-2.5.2 (Reference 2.1-8). Note the BWRX-300 Small Modular Reactor (SMR) is built within the DNNP site boundary with a smaller footprint of approximately 9,800 m<sup>2</sup>, per the 2022 NK054-REP-01210-00142 (Reference 2.1-5), compared with the original application involving much larger nuclear power plants, per the 2010 NK054-REP-01200-10000 (Reference 2.1-6).

Chapter 15, Section 15.7 includes tabulated summaries listing the DSA results for bounding BWRX-300 AOO and DBA event sequences. Also, Chapter 15, Appendix 15A demonstrates implementation of the D-in-D provisions ensures protection against unacceptable radiation releases. Chapter 15, Section 15.7 thus concludes all BWRX-300 analyzed bounding AOOs, DBAs or DECAs without core damage have met the dose acceptance criteria for the 350 m exclusion zone.

#### **2.1.2.2 Security Requirements**

The security requirements for the DNNP site and the BWRX-300 facility and how such security requirements are met are described in the Security Annex, which is an OPG Confidential Protected Security document.

#### **2.1.2.3 Description of Site Layout**

The high-level description of the DNNP site layout includes:

- The Power Block that encompasses several buildings and a plant services area (refer to Chapter 1, Figure A1.5-1)
- Locations of the site vehicle entrance (sally port) as well as roads to allow access of trucks and individuals to Power Block buildings, with the Protected Area Access Building located west of the sally port (refer to Chapter 1, Figure A1.4-1)
- Locations of the irradiated fuel dry storage (which is regulated under a separate licence), Pumphouse/Forebay, intake shaft and tunnel, discharge structure and tunnel, and switchyard and transmission lines (refer to Chapter 1, Figure A1.1-2)
- Heavy haul routes for the construction phase of Unit 1 as shown in Chapter 1, Figure A1.1-2, and for the construction phases of Units 2, 3, and 4, as shown in Figure 5 of the 2022 EIA (Reference 2.1-4).

#### **2.1.2.4 Minimizing Environmental Impacts**

Measures are included in the DNNP site layout and BWRX-300 design to minimize the impact on the surrounding region and the environment, per the 2022 NK054-REP-07730-00055 (Reference 2.1-4), for example:

1. The location and placement of the lakebed intake structure regarding the commitment for fish entrainment and impingement as well as the discharge diffusers to meet the commitment for effluent plume in the 2021 NK054-REP-01210-00078 (Reference 2.1-2)
2. Consideration of sensitive land features, such as shoreline bluffs and Bank Swallows, habitat to the extent practicable
3. A smaller BWRX-300 footprint which does not need any additional land area that could be obtained from lake infill
4. Designing into the site storm water management provisions for the construction and post construction phases

5. Minimizing the area of disturbance for permanent structures as well as the areas for spoils on the DNNP site by optimizing the BWRX-300 footprint
6. Cooling towers are not used for the BWRX-300 for either the normal or ultimate heat sinks, per Table 3 of the 2022 EIS NK054-REP-07730-00055 (Reference 2.1-4); thus, the adverse effects associated with cooling towers (e.g., effects on the visual landscape and socio-economic conditions) are not applicable
7. The primary and secondary heat transport systems are combined, and use is made of natural circulation and passive safety systems resulting in an optimized size of the facility and contributing to lowering the risk of normal and abnormal operating conditions

### **2.1.3 Population Distribution and Density**

The Municipality of Clarington and the City of Oshawa have both experienced steady growth over the last ten years.

According to recently released Statistics Canada data, Clarington's population was 101,427 in 2021, which is an increase of 10.2% from that in 2016 when the population was recorded at 92,130. The rural area of Clarington had a population of 11,297 in 2021. The Municipality of Clarington Official Plan forecasts that Clarington will have a population of 140,340 by 2031, with 124,685 in its urban areas and 15,655 in its rural areas. The 2021 population data listed in Table 2.1-2 for the Municipality of Clarington is distributed amongst four urban areas including Courtice, Bowmanville, Orono, and Newcastle as shown in Figure 2.1.3-1.

**Table 2.1-2: Population Data for the Municipality of Clarington for 2021**

Urban Area	Population
Courtice	28,545
Bowmanville	47,176
Orono	2,476
Newcastle	11,933
Total	90,130

The population of the City of Oshawa was 149,607 in 2011 and grew to 159,458 in 2016, which was a 6.6% increase. The City of Oshawa's Official Plan provides population forecasts of 174,695 in 2021, 184,460 in 2026 and 197,000 in 2031.

Refer to Subsection 2.8.4 for detailed 2016 population data that is broken into sectors by distance and direction for use in air dispersion modeling within a 30 km radius of the Darlington Nuclear site.

### **2.1.4 Municipal Services**

Within the 10 km survey area, there are 17 education institutions available for students: 12 primary schools and five secondary schools. As well, there are six fire emergency stations (excluding OPG's on-site Darlington fire station) and one regional police station (plus one administrative police department). Additionally, there is one hospital - Lakeridge Health in Bowmanville.

### **2.1.5 Site Access and Transportation Networks**

The Darlington Nuclear site can be accessed via two roads. Holt Road runs north to south and allows for direct access to the site. Energy Drive runs west to east and connects to Park Road for access to the site. Multiple parking lots are present on the site.

Within 10 km of the site, there are many arterial roads, minor arterial roads, highways, residential roads, and rural roads. These roads fall within the borders of the 10 km survey area defined in Subsection 2.1.1. A complete list of roads falling within the surveyed area can be found in Appendix B.

Transportation networks of significance are listed in the following:

1. Three 400-series highways are located within 10 km of the site - Highways 401, 407, and 418 (refer to Subsection 2.3.1(b) for supplementary information on Highway 401).
2. Two railway lines are located within 10 km of the site which converge and run adjacent to one another east of Lakeshore Road, Newcastle:
  - a. The Canadian Pacific line runs west east, which is located just north of Highway 401, and is used for trains transporting cargo.
  - b. The Canadian National line runs west east, which is located south of Highway 401 and used for trains transporting people and cargo, and part of which bisects the DNNP and DNGS sites (refer to Subsection 2.3.1 for further information, and Subsection 2.2.3.2(a) for hazards related to potential railway accidents).
3. Oshawa Executive Airport is located at the southeast corner of Taunton Road and Thornton Road North. The airport is located just outside the 10 km survey area (refer to Subsection 2.3.1(c) for additional information).
4. The Port of Oshawa East Pier (at the bottom of Simcoe Street South) is located west of the site and allows cargo ships to receive/deliver shipments.
5. St. Marys Cement has a private dock at its facility to the east of the DNNP site for the shipment of aggregate from its operations.

### **2.1.6 Public Transit**

The closest regional transit stop to the site is located at Old Holt Road and King Street, approximately 5 km north of the site. The stop is part of the 902A King bus line offered by Durham Regional Transit and runs west east through the Durham Region. Additionally, the region introduced two types of on-demand transportation services in the Durham Transportation Master Plan (2017): transit on-demand and rural on-demand. Transit on-demand allows riders to request a ride with pickup located at their nearest transit stop, while rural on-demand allows riders to request a ride with pickup at their current location. The region also has a park and ride station within the survey area located at Courtice Road north of Highway 401.

The closest transit stop to the site is a GO Bus stop located at Bowmanville Avenue and Baseline Road. The stop is part of the 88 GO Bus Route that is running from Oshawa to Peterborough with multiple bus stops located north of the site along Bowmanville Avenue and King Street. Additionally, there are two proposed GO Transit stations within the survey area. GO Transit's Lakeshore East Rail Service will operate on the Canadian Pacific rail line north of Highway 401, which will include service to the two proposed stations: Courtice GO (Courtice Road north of Baseline Road) and Bowmanville GO (Bowmanville Avenue north of Aspen Spring Drive). Per correspondence with Durham Region staff, the Courtice and Bowmanville GO stations are



projected to be operational in 2026. Furthermore, two secondary plans are currently being developed for the areas adjacent for each proposed GO station.

### **2.1.7 Active Hiking and Cycling Trails**

As shown in Figure 2.1.8-1, the Darlington Waterfront Trail, part of the Great Lakes Waterfront Trail, is a multi-use path that forms part of the recently approved Durham Regional Cycling Plan. The trail is used by pedestrians and cyclists for transportation or recreational purposes, provides direct access to the Darlington Nuclear site and falls within OPG owned lands. Additionally, hiking trails are available near Lakeview Park in Oshawa, as the Larry Ladd Harbour Trail connects to Lakeview Beach. The Primary Cycling Network Durham currently provides over 400 km of cycling infrastructure in the region.

### **2.1.8 Park Spaces and Waterbodies**

There is abundance of parks, greenspaces, conservation areas, and waterbodies located within the 10 km survey area, with multiple public recreational spaces directly adjacent to Darlington Nuclear site. As detailed in Subsection 2.1.7, part of the Darlington Waterfront Trail runs through the Darlington Nuclear site. Directly adjacent to the west of the DNGS site is Alijco Beach, a beachfront which can be accessed by users for recreational purposes. Other park spaces and waterbodies are dispersed throughout the rest of the survey area, with places of significance listed below:

1. One provincial park falls within the survey area - Darlington Provincial Park.
2. The Darlington Hydro Soccer Fields facility (owned by OPG and licensed to the Municipality of Clarington) falls within the survey area, as does Bowmanville's Baseball Fields Complex (located at Green Road just north of Highway 401).
3. Five conservation areas fall within the survey area: three are located in Bowmanville (Bowmanville Valley Conservation Area, Bowmanville Westside Conservation Area, Stephen Gulch's Conservation Area) and two are located in Oshawa (Harmony Valley Conservation Area, Oshawa Valleylands Conservation Area).
4. Three beaches fall within the survey area: two are located in Bowmanville (Alijco Beach, Port Darlington Beach) and one is located in Oshawa (Lakeview Beach).

A complete list of park spaces and water bodies falling within the surveyed area can be found in Appendix C.

### **2.1.9 Industrial Facilities**

The industrial facilities that are within the survey area of 10 km and directly adjacent to Darlington Nuclear site are discussed in Subsection 2.1.1.

Other industrial facilities are dispersed throughout the rest of the survey area, with most facilities located west of the site in Oshawa. A complete list of industrial facilities falling within the surveyed area is found in Appendix A.

While not located in the survey area, the PNGS is located approximately 25 km west of the Darlington Nuclear site (refer to Subsection 2.1.1 and Subsection 2.2.5.2).

### **2.1.10 References**

- 2.1-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.1-2 NK054-REP-01210-00078 R007, 2021, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- 2.1-3 NK054-CORR-00531-10635, 2022, "DNNP: Submission of Environmental Assessment Follow-Up Monitoring Plans / Methodology Reports and Request for Acceptance and Closure of Their Respective Commitments under D-P-12," Ontario Power Generation.
- 2.1-4 NK054-REP-07730-00055-R000, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report for Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.1-5 NK054-REP-01210-00142-R000, 2022, "Darlington New Nuclear Project – Site Evaluation Update Summary Report," Ontario Power Generation.
- 2.1-6 NK054-REP-01200-10000 R005, 2010, "Use of Plant Parameters Envelope to Encompass the Reactor Designs being considered for the Darlington Site," Ontario Power Generation.
- 2.1-7 Durham Region, "Courtice Water Pollution Control Plant - 2021 Annual Performance Report."
- 2.1-8 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.1-9 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



Figure 2.1.1-1: Darlington Nuclear Site Proximity to Pickering Nuclear Generating Station



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



Figure 2.1.1-2: Darlington Nuclear Generation Station and Darlington New Nuclear Project Lands



Figure 2.1.1-3: Darling Nuclear Generating Station and – Darlington New Nuclear Project Proximity to Industry



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

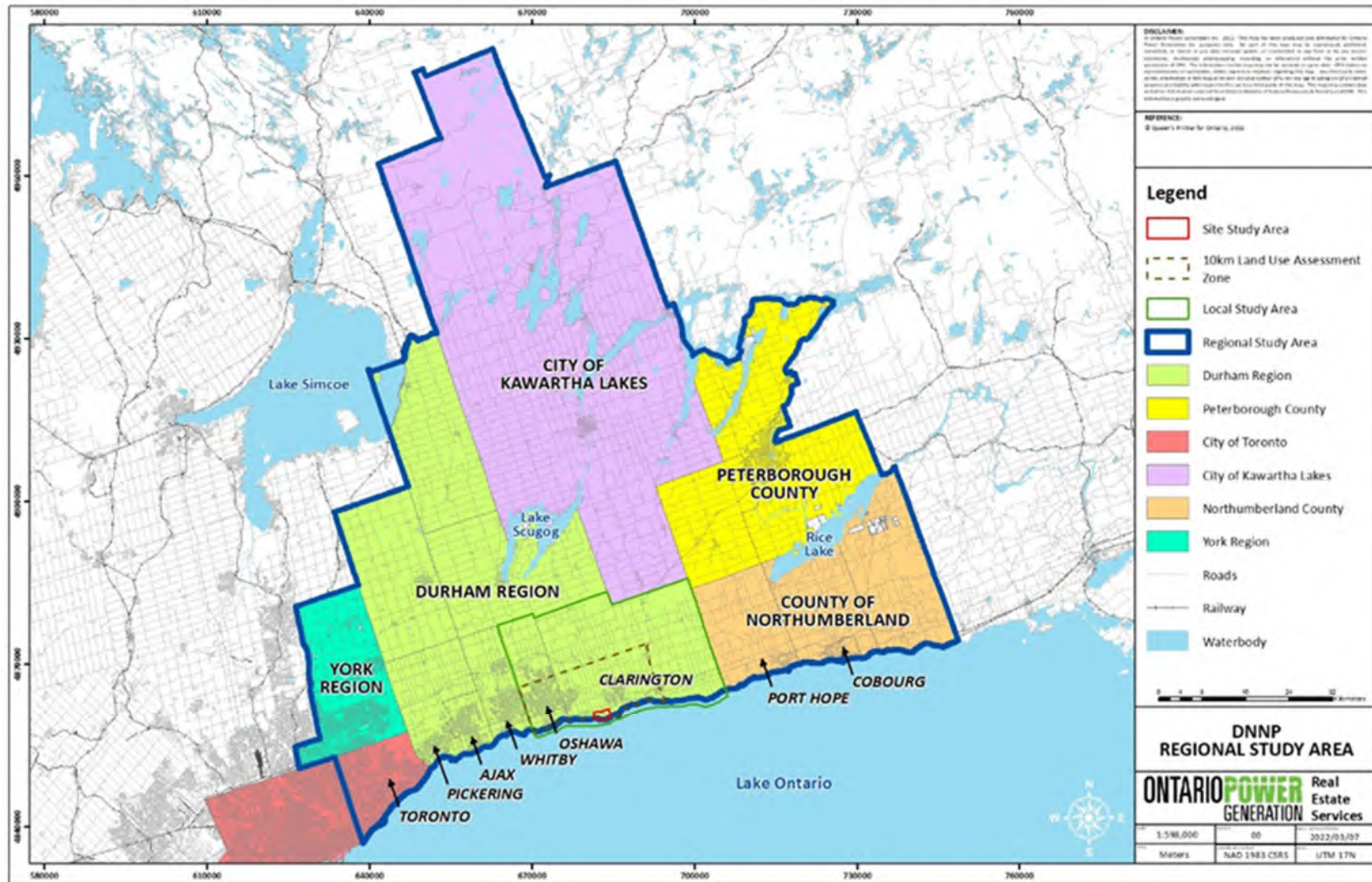


Figure 2.1.1-4: DNNP Regional Study Area

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

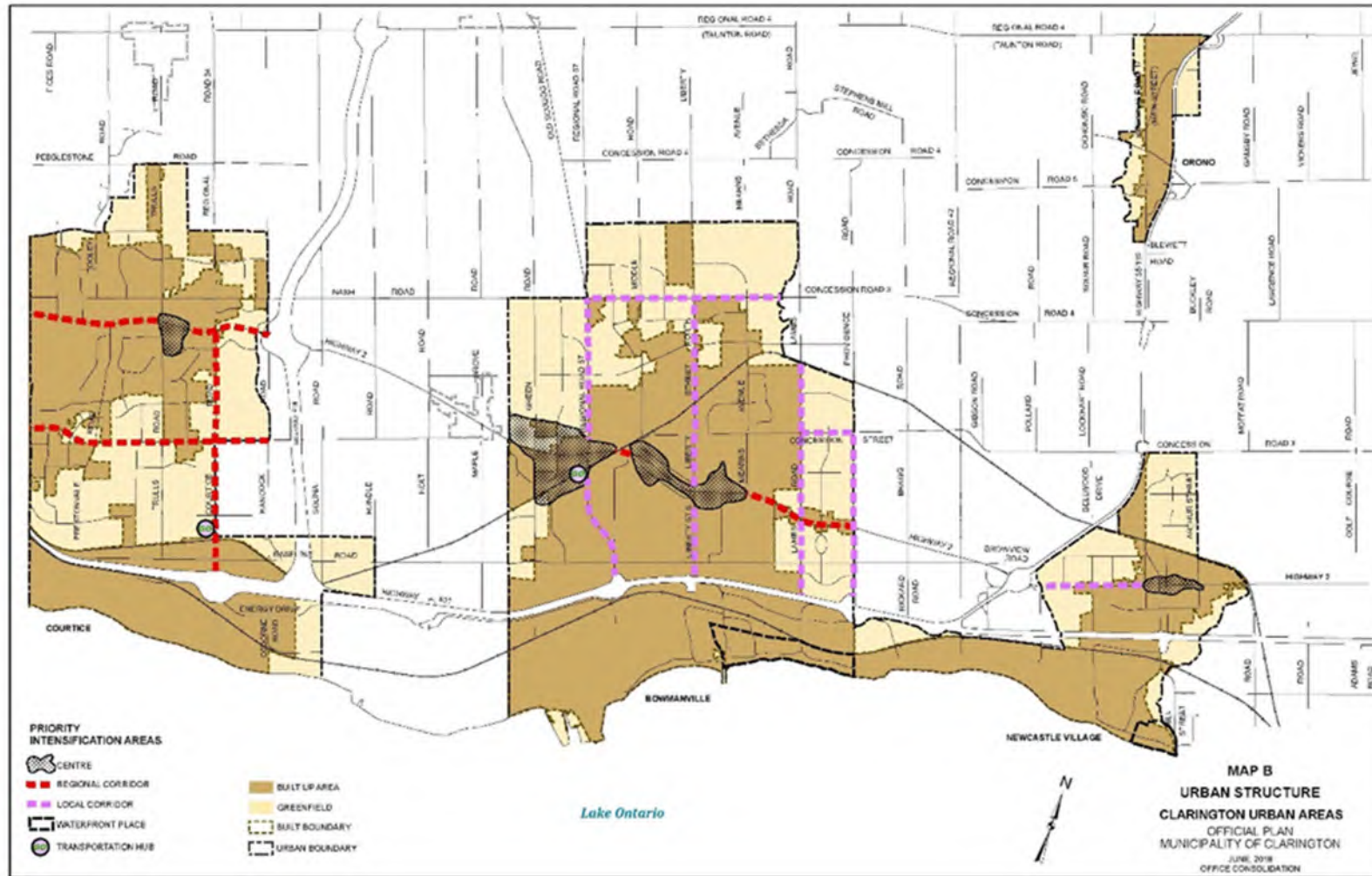


Figure 2.1.3-1: Clarington Urban Areas



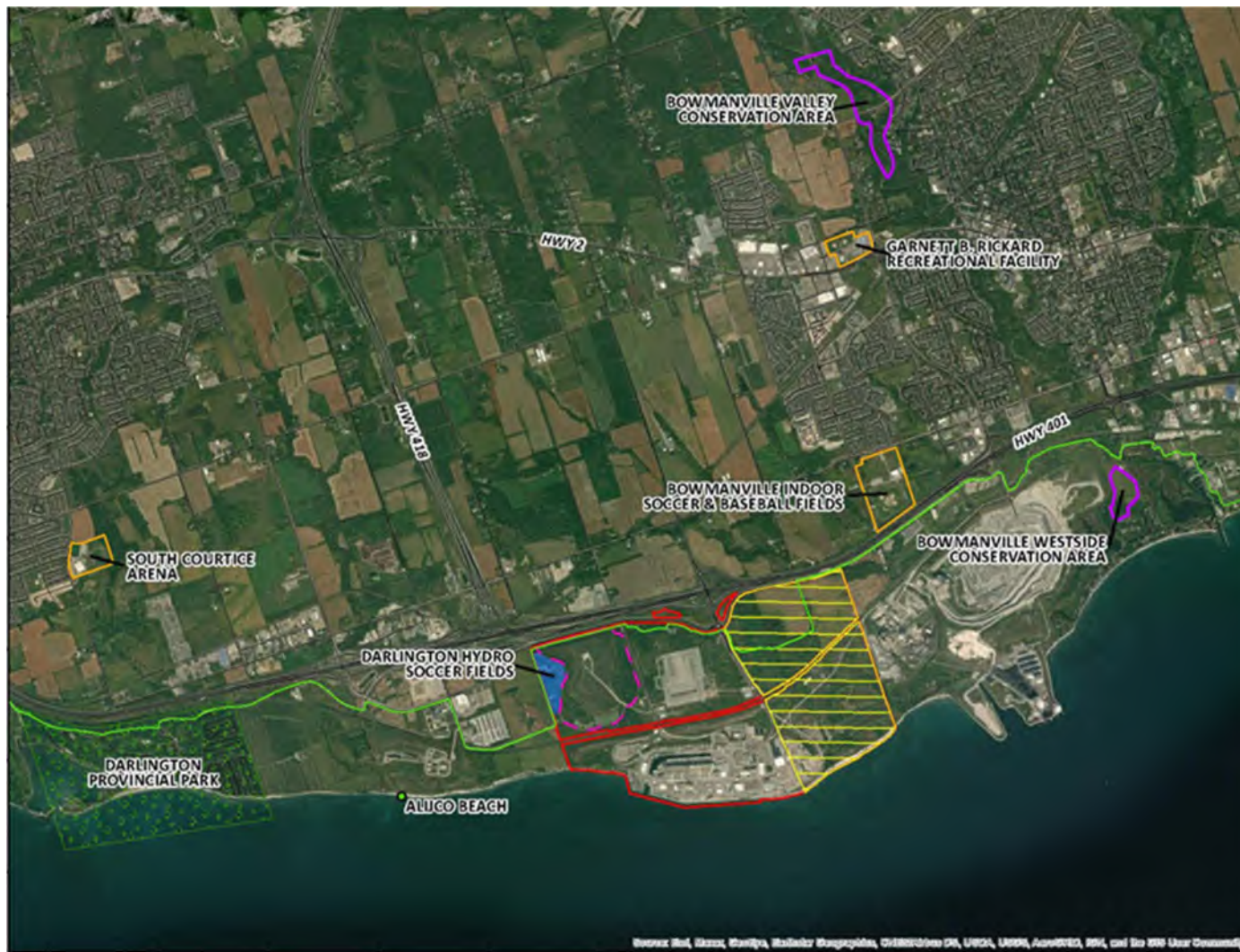


Figure 2.1.8-1: Darlington Nuclear Site – Active Darlington Waterfront Trail



## **2.2 Evaluation of Site-Specific Hazards**

Section 2.2 characterizes and quantifies site-specific hazards that are used in the design of the BWRX-300 and builds upon the 2022 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10). As the DNNP and DNGS share the Darlington Nuclear site (refer to Subsection 2.1.1), the DNGS 2019 Hazard Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) is used in support of Section 2.2 and to inform the DNNP hazard screening analysis. All such site characteristics are validated for the BWRX-300 Unit 1 design and its location on the DNNP site, as shown in Chapter 1, Figure A1.1-2.

### **2.2.1 Introduction**

Section 2.2 includes the methodology used for and the results of the evaluation of site-specific external hazards associated with the DNNP site and the BWRX-300 facility. Such evaluation is derived from previous DNNP hazards assessment work completed in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) and the 2009 NK054-REP-01210-00019 (Reference 2.2-2) as well as from a 2019 DNNP site preparation licence renewal activity report NK054-REP-01210-00108 (Reference 2.2-3). The evaluation addresses specific items relevant to DNNP site-specific external hazards, as identified in the 2020 OPG's application to renew the DNNP site preparation licence NK054-CORR-00531-10533 (Reference 2.2-4).

The methodology used to evaluate external hazards is described in Subsection 2.2.2.

The hazards identified for further evaluation are:

- Subsection 2.2.3: Transportation Accidents, Including Toxic Chemical or Gas Releases / Explosions Hazards
- Subsection 2.2.4: Stationary Non-nuclear Accidents Hazards
- Subsection 2.2.5: Stationary Nuclear Accidents Hazards
- Subsection 2.2.6: Industrial Hazards
- Subsection 2.2.7: Biological, Animal, and Frazil Ice Hazards
- Subsection 2.2.8: Ice Storm Hazard
- Subsection 2.2.9: Electromagnetic Interference Hazard
- Subsection 2.2.10: On-site Methane Hazard

A summary results and follow-up considerations of the hazards listed above are provided in Table 2.2-1.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.2-1: Screening and Validation of CNSC–Identified DNNP  
Site-Specific Hazards**

2.2.2 External Hazards Evaluation Methodology			
Methodology	The methodology and criteria used in the 2019 DNGS NK38-REP-03611-10043 (Reference 2.2-5) Comparable methodology and criteria developed in the 2022 BWRX-300 DNNP NK054-REP-01210-00144 (Reference 2.2-10)		
Screening Criteria	Qualitative Criteria – QL-1 to QL-7 Quantitative criteria – QN-1 to QN-5		
2.2.3 Characterization of Hazards from Transportation Accidents, Including Toxic Chemicals or Gas Releases/Explosions			
2.2.3.1 Hazards from Air Transportation Accidents			
Small aircraft	Screened out	QL-1: Equal or lesser damage than similar design basis event	The small aircraft crash is screened out as the BWRX-300 is designed to withstand site-specific automobile tornado missiles, per Subsection 2.6.6.
Large military aircraft	Screened out	QL-3: Cannot occur at or close enough to the site to affect BWRX-300	Large bombers, large cargo planes, fuel tankers, or heavily armed jet fighters do not fly in the vicinity of the Bowmanville airspace
Large civil aircraft	Screened out	QN-5: Frequency of <1.0E-7/yr	NOTE: Malevolent large aircraft crash is analyzed in the Security Annex.
2.2.3.2 Characterization of Hazards from Rail Transportation Accidents			
Release of toxic gases	Screened in as DEC	Hazard frequency is estimated at 1.9E-06 occ./yr. Thus, this hazard is a Beyond Design Basis Accident (BDBA) DEC, as documented in NK054-REP-01210-00150 (Reference 2.2-11)	
Explosions	Screened in as DEC	Hazard frequency is estimated at 9.0E-07 acc./yr Thus, this hazard is a BDBA DEC, as documented in NK054-REP-01210-00149 (Reference 2.2-12)	
2.2.3.3 Characterization of Hazards from Road Transportation and Traffic Accidents			
Release of toxic or asphyxiant material	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the Darlington Nuclear site is about 1.0 km away from Highway 401.
2.2.3.4 Characterization of Marine Transportation			
Chemical Leak	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant QL6: Does not cause an initiating event	Commercial shipping is approximately 27 km away for the DNNP. The consequence of a chemical leak from a tanker or a cargo ship, would be mostly an environmental hazard, and would not have an impact on safe operation of the station.
Release of toxic gases	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the DNNP is about 27 km away from the general tanker or cargo ship commercial routes in Lake Ontario.
Explosion	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the DNNP is about 27 km away from the general tanker or cargo ship commercial routes in Lake Ontario.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Physical Damage	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant  QL1: Bounded by the impact of damage caused by frazil ice described in Subsection 2.2.7.2	Hazards from accidents involving recreational boats or vessels pose no significant threat to the BWRX-300 safe operation, even if the accidents occur near the lake water intake structure.  Also, a restricted zone is established around the BWRX-300 offshore structures.
<b>2.2.4 Characterization of Stationary Non-Nuclear Accidents</b>			
2.2.4 Fire – Natural Gas Pipelines	Screened out	QL6: Does not cause an initiating event or relevant safety function	There are no substantial pipelines carrying large quantities of natural gas, close enough to the site.
2.2.4.1 Release of toxic gases or chemical from commercial outlets in the area	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant  QL5: The event is slow to develop so there is sufficient time to eliminate the source of adequately respond	There are no industrial toxic gas or chemical storage tanks or pipelines carrying significant quantities of natural gas close enough to the site.  Assumed St. Marys toxic release is not close enough to the site to affect the plant
2.2.4.2 Explosion – Shock Waves	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	Distances between DNNP and: <ul style="list-style-type: none"><li>• Cigas Propane tanks are about 3.6 km far from the DNNP site</li><li>• St. Marys diesel fuel tanks is greater than 700 m from the Power Block of multi-unit layout (Reference 2.2-16).</li></ul>
2.2.4.2 Explosion - Missiles Hydrogen used for Tritium Removal Facility	Screened out	QL3: Large Missiles - Cannot occur on or close enough to the site to affect the plant  QL4: Small Missiles - Bounded by design basis tornado in Subsection 2.6.6	The Tritium Removal Facility is located approximately 1.0 km west of the DNGS vacuum building.
<b>2.2.5 Characterization of Stationary Nuclear Accidents Hazards</b>			
2.2.5.1 Cameco's Port Hope Uranium Conversion Facility	Screened out	The facility is located on the north shore of Lake Ontario, approximately 40 km east of Darlington Nuclear site. The Cameco plant is a chemical processing facility with negligible radioactive releases.	
2.2.5.2 PNGS	Screened out	Any hazard from PNGS irradiated fuel still within an irradiated fuel bay or a dry storage facility is bounded by the much closer event from DNGS. Based on (Reference 2.2-5), PNGS radioactive release event is characterized as a slow developing event, allowing sufficient time for operators to take appropriate actions (if warranted), and can therefore be screened out.	
2.2.5.3 DNGS – Exclusion Zone	Screened in	The DNNP site is partly within the exclusion zone of DNGS.	
<b>2.2.5.4 Characterization of Other Radiological Hazards from DNGS</b>			
2.2.5.4.1 DNGS – Tritium Removal Facility – Tritium Release	Screened out	Evaluations in (Reference 2.2-1) and (Reference 2.2-5) determined that regulatory dose limits at the site boundary apply to all these nuclear events with negligible impact to DNNP.	

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.2.5.4.2 DNGS – Irradiated Wet Fuel Storage Facility	Screened out		
2.2.5.4.3 DNGS – Irradiated Dry Fuel Storage Facility	Screened out		
2.2.5.4.4 DNGS – Radioactive Waste Storage	Screened out		
2.2.6 Characterization Industrial Hazards (St. Marys)			
St. Marys Cement Plant – Uncontrolled blasts	Screened in	St. Marys Cement commits to carry out blasts with a maximum allowable horizontal, vertical, longitudinal, and radial velocities of less than 3 mm/s measured at the Darlington Nuclear site property boundary with St. Marys.	
2.2.7 Characterization of Biological, Animal and Frazil Ice Hazards			
2.2.7.1 Water-based Biological	Screened out	QL4: Bounded by the impact of damage caused by frazil ice described in Subsection 2.2.7.2	Hazards associated with blockage of intake cooling water resulting in the loss of heat sink
2.2.7.1 Airborne birds or insects	Screened out	QL-1: Equal or lesser damage than similar design basis event	This event is equivalent to outside air damper isolation during off-normal conditions
2.2.7.2 Frazil Ice	Screened in	Frazil ice is considered a potential hazard for causing water intake blockage to DNNP.	
2.2.8 Characterization of Ice Storm Hazard			
Ice Storm	Screened out	QL-1: Equal or lesser damage than similar design basis event	For the DNNP BWRX-300, the loss of the switchyard is part of the Loss-of-Preferred Power (LOPP), an Anticipated Operational Occurrence, which is the Pressure Increase Group and is designated as a BL-AOO event
2.2.9 Characterization of Electromagnetic Interference Hazard			
Electromagnetic Interference	Screened in	Since electromagnetic interference sources (e.g., high-voltage transmission lines and communication towers) are continuously present, the risk of electromagnetic interference at the site must be addressed in the design basis of the BWRX-300	
2.2.10 Characterization of On-site Methane Hazard			
During construction	Screened in	Methane gas is harmful to the health of humans and is combustible. Methane gas must be monitored during excavation, especially for the RB, since the methane is expected to dissipate quicker than what was observed in the boreholes due to the significantly larger air space.	
Post construction	Screened in	Methane in bedrock during operation is added as a hazard to be considered during design	

### **2.2.2 External Hazards Evaluation Methodology**

The 2019 Hazards Screening Analysis reported in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) provides a comprehensive assessment of the hazards associated with the DNGS site. Given that the DNNP site is within the Darlington Nuclear site (refer to Chapter 1, Figure A1.1-2) and in geographic proximity with the DNGS site, this analysis is deemed applicable to support and inform the evaluation of the external hazards listed in Subsection 2.2.1 for the DNNP site. In addition, since the DNGS external hazard screening methodology NK38-REP-03611-10043 (Reference 2.2-5) is aligned with the 2022 BWRX-300 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10), the results of the 2019 DNGS analysis in NK38-REP-03611-10043 (Reference 2.2-5) are used to supplement and validate the DNNP site-specific external hazards evaluation reported in the 2022 DNNP NK054-REP-01210-00144 (Reference 2.2-10).

In the 2019 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00108 (Reference 2.2-3), detailed DSA and PSA are performed during the BWRX-300 design phase. The DSA and PSA updates are performed in compliance with CNSC REGDOC-2.4.1 (Reference 2.2-14) and REGDOC-2.4.2 (Reference 2.2-15), respectively, and are tracked under the 2021 DNNP Commitment D-C-3 NK054-REP-01210-00078 (Reference 2.2-8). With respect to external hazards, DNNP Commitment D-C-3 also requires “the design of the new plant must demonstrate that it can mitigate the identified hazards to ensure that the required safety goals are met.”

The screening methodology and criteria used to assess hazards are described is found in Section 1.0 of the DNGS 2019 NK38-REP-03611-10043 (Reference 2.2-5). The screening technique involved a systematic approach starting with a qualitative assessment of the impacts of hazards on the safe operation of the station, followed by a quantitative screening of hazards not being screened out qualitatively. The methodology follows OPG’s PSA guides for screening of internal and external hazards.

The 2022 BWRX-300 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10) builds on the 2019 Darlington screening technique NK38-REP-03611-10043 (Reference 2.2-5) and devises comparable criteria for the BWRX-300 facility. The developed qualitative and quantitative screening criteria are applicable to screening internal and external hazards, as listed in Appendix B of the 2022 NK054-REP-01210-00144 (Reference 2.2-10).

The following criteria are used for qualitative screening of hazards in the 2019 NK38-REP-03611-10043 (Reference 2.2-5):

- QL-1: The event is of equal or lesser damage potential than similar events for which the plant has been designed.
- QL-2: The event has a significantly lower reactor sources likelihood than another event that has been screened out, and yet the event could not result in worse consequences than the other event.
- QL-3: The event cannot occur at the site or close enough to the site to affect the plant.
- QL-4: The event is included in the definition of another event.
- QL-5: The event is slow in developing such that it can be demonstrated that there is sufficient time to eliminate the source of the threat or provide an adequate response.
- QL-6: The event does not cause an initiating event (including the need for a controlled shutdown) as well as safety system function losses needed for the event.
- QL-7: The consequences to the plant do not require the actuation of front-line systems.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

NOTE: QL-1 to QL-5 apply to both the reactor and non-reactor sources. QL-6 and QL-7 apply only to reactor sources and not to the non-reactor sources.

The following criteria are used for quantitative screening in the 2019 NK38-REP-03611-10043 (Reference 2.2-5).

- QN-1: Severe Core Damage Frequency  $< 1.0E-6/\text{yr}$ . Applies only to reactor sources and not to non-reactor sources.
- QN-2: Design Basis Hazard Frequency,  $< 1.0E-5/\text{yr}$  and Conditional Core Damage Probability  $< 0.1$ . Applies to reactor sources only and not to non-reactor sources.
- QN-3: Severe Core Damage Frequency  $< 10^{-7}/\text{yr}$ . Applies to the reactor sources only. An equivalent QN for non-reactor sources of Low Release Frequency (LRF)  $< 1.0E-7/\text{yr}$  is considered.
- QN-4: Design Basis Hazard Frequency,  $< 1.0E-6/\text{yr}$  and Conditional Core Damage Probability  $< 0.1$ . Applies to reactor sources only. An equivalent QN for non-reactor sources is considered as follows: Design Basis Hazard Frequency,  $< 1.0E-6/\text{yr}$  and conditional large release probability (CLRP)  $< 0.1$ .
- QN-5: Initiating Event or Hazard Frequency may be screened out if it can be shown that their frequency is  $< 1.0E-7/\text{yr}$ . Applies to both reactor and non-reactor sources.

The application of this methodology results in hazards being “screened out” or “screened in.” “Screened out” implies that the hazard does not pose any safety concerns. “Screened in” implies further assessment is required to address the hazards. Hazards which are neither qualitatively nor quantitatively screened out, are addressed during detailed Probabilistic Safety Assessments (for example, seismic, high winds).

### **2.2.3 Characterization of Hazards from Transportation Accidents, Including Toxic Chemicals or Gas Releases/Explosions**

Evaluations of hazards from transportation accidents are detailed as follows:

- By air - Subsection 2.2.3.1
- By train - Subsection 2.2.3.2
- By road - Subsection 2.2.3.3
- By marine – Subsection 2.2.3.4

Previous assessment results for DNNP hazards associated with transportation events are provided in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) and the 2009 NK054-REP-01210-00019 (Reference 2.2-2). The evaluations presented in Subsection 2.2.3 address the specific issues identified by the CNSC in Subsections 4.6.1, 4.6.2 and 4.6.3 of the 2020 Renewal Application for DNNP Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.2-4).

Aircraft crashes and ship accidents were evaluated for the DNNP site in Section 4.3 and 4.4 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). The evaluation did not consider the impact from toxic chemicals or gas releases/explosions specific to these accidents. However, the impact from toxic chemicals or gas releases/explosions from transportation accidents were implicitly assessed in Section 4.6 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1).

Further, Section 4.6 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) evaluated the risks associated with hazardous fluids, including toxic clouds from the release of toxic gases, deflagrations (explosions) from the release of liquified petroleum gases and flammable pressure

liquified gases. The evaluation determined toxic gas clouds reaching the DNNP site at high enough concentrations have the potential to impact the Main Control Room (MCR) and Secondary Control Room (SCR) habitability of the proposed plant (that is, the BWRX-300 nuclear facility). Refer to Chapter 6, Section 6.4 for further details on habitability of the MCR and SCR.

With respect to explosions, the evaluation in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) identified potential damage to buildings from missiles resulting from Boiling Liquid Expanding Vapour Explosion (i.e., tanks containing liquified petroleum); when travelling at high velocity, these missiles can damage outdoor and indoor equipment. The evaluation determined that the overpressure effects due to explosion on the building must be mitigated. Mitigation may require the use of an appropriate physical barrier or the physical separation of important safety equipment/systems. The evaluation stated that requirements for this hazard is to be considered during the detailed design phase of the project (that is, BWRX-300). The 2019 DNGS Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) also assessed the release of toxic chemicals and gas/release explosions from transportation accidents. The data used for DNGS hazards analysis supplement the DNNP site-specific data that are employed in the design and safety analysis stage of the DNNP BWRX-300, as applicable.

For additional information specific to toxic gas and chemical hazards, refer to Subsection 2.2.3.2 for rail transportation accident hazards, Subsection 2.2.4.1 for release from stationary hazards, and Subsection 2.4.1 for on-site hazards.

#### **2.2.3.1 Characterization of Hazards from Air Transportation Accidents**

Two types of aircraft are examined: small and large (both civil and military).

1. The small aircraft crash is screened out qualitatively as not having an impact on the safe operation of the facility, based on the screening criterion QL1 that the event is of equal or lesser damage potential than similar events for which the plant is designed. Per Section 3.1 of the 2019 Darlington hazard screening analysis (Reference 2.2-5), small aircraft impact is bounded by tornado missiles. The small aircraft crash is therefore screened out as the BWRX-300 will be designed to withstand automobile tornadoes missiles (refer to Subsection 2.6.6).
2. Large aircraft (military) aviation accidents are not a concern for the Darlington Nuclear site, as there are no large bombers, large cargo planes or fuel tankers, or heavily armed jet fighters flying in the vicinity of the Bowmanville airspace, per the 2020 NK054-CORR-00531-10533 (Reference 2.2-4).
3. Large aircraft (civil) accidents are screened out under screening criterion QN5 (refer to Subsection 2.2.2) based on a hazard frequency of  $<1.0E-7/\text{yr}$ .

#### **2.2.3.2 Characterization of Hazards from Rail Transportation Accidents**

As described in Subsection 2.1.5, two railway lines run within the 10 km study area surrounding the Darlington Nuclear site. Of particular relevance is the Canadian National Railway line which bisects the Darlington Nuclear site and passes approximately 600 m north of the DNNP site. This railway line has potential hazards associated with assumed derailment accidents involving one or more cargo cars.

Rail transportation accidents are assessed in the 2019 DNGS Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5), the 2022 DNNP Rail Transportation – Toxic Gas/Chemical Release Hazard Assessment NK054-REP-01210-00150 (Reference 2.2-11), and the 2022 DNNP Rail Transportation – Explosion Hazard Assessment NK054-REP-01210-00149 (Reference 2.2-12). The objective is to address hazards associated with train derailment and crash, including

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

cold or hot toxic gas releases, as well as Vapour Cloud Explosions, Boiling Liquid Expanding Vapour Explosion, and other types of explosions.

The assessments considered the two rail lines running “east-west” directly north of Darlington Nuclear site. Of particular interest is the Canadian National Railway Toronto to Montreal main line which passes through the OPG DNGS and DNNP sites, as shown in Figure 2.1.1-2.

One of the hazards analyzed in the 2022 NK054-REP-01210-00150 (Reference 2.2-11) is the possibility of a large toxic gas/chemical release. A consequential harm from this hazard could be a toxic gas/chemical release that would be airborne toward the DNNP site with the capacity for widespread and distant impact. Another hazard is the potential of large explosion, analyzed in NK054-REP-01210-00149 (Reference 2.2-12), involving explosive commodities being transported by the railway line, occurring in the vicinity of DNNP BWRX-300 structures and components. The following toxic gas release and explosion scenarios are assessed in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the 2022 NK054-REP-01210-00150 (Reference 2.2-11), and the 2022 NK054-REP-01210-00149 (Reference 2.2-12), for applicability to DNNP:

1. Cold Toxic Gases Release: Release and dispersion of airborne toxic chemicals or asphyxiants toward the BWRX-300 HVAC intakes that could expose the station staff to toxic chemicals and result in challenging the habitability of work areas.
2. Hot Toxic Gas Release: Similar to cold toxic gas releases, if the train derailment accident involves fire, it could result in hot toxic gas releases. Combustible chemicals could result in releasing an intense heat, causing secondary combustion of other materials (e.g., insulations, containers and covers), and such releases usually involve other chemicals that can have a wide range of toxicities. Heavy hydrocarbons produce a significant amount of carbon dioxide, carbon monoxide and soot when they catch fire. Some chemicals may produce toxic byproducts while burning, such as hydrazine (combustion byproducts include nitrogen dioxide, which is highly toxic).
3. Hydrocarbon Explosions: Release of light hydrocarbons with high vapour pressures (flammable), when transported under high pressure (e.g., liquefied petroleum gas), can produce two types of explosions:
  - a. Boiling Liquid Expanding Vapour Explosion: Boiling Liquid Expanding Vapour Explosions could generate missiles, fireballs, and blast waves. Missiles could travel hundreds of meters from the source. Blast waves from Boiling Liquid Expanding Vapour Explosions are normally localized.
  - b. Vapour Cloud Explosion: With Vapour Cloud Explosions, vapour cloud ignition is delayed after the cloud has dispersed somewhat and mixed with air. Vapour Cloud Explosions produce blast waves that could damage buildings and equipment.
  - c. Confined Explosions: A flammable fluid can produce a confined explosion if it becomes airborne, mixes with air, and is ignited in a confined space. This would produce a so-called Confined Explosion. Such an explosion could arise in a building, a room, or the vapour space of a storage tank. Blast waves from confined explosions are localized.

The hazard from the release of toxic gases resulting from Canadian National Railway assumed transportation accidents close to the DNNP site have an estimated frequency of 1.9E-06 occ./yr, per the 2022 NK054-REP-01210-00150 (Reference 2.2-11). Thus, it is screened out from design basis input since it is assessed as a Beyond Design Accident (BDBA) DEC, per REGDOC-2.4.1 (Reference 2.2-14).



Similarly, the explosion hazard from a Canadian National Railway derailment accident near the DNNP site has an estimated frequency of  $9.0\text{E-}07$  occ./yr, per the 2022 NK054-REP-01210-00149 (Reference 2.2-12). Consequently, it is screened out from design basis input based on the assessment that it is a BDBA DEC, per REGDOC-2.4.1 (Reference 2.2-14).

#### **2.2.3.3 Characterization of Hazards from Road Transportation and Traffic Accidents**

Road transportation and traffic accidents are assessed in 2019, and results for DNGS are reported in Subsection 3.2.3 of per the 2019 NK38-REP-03611-10043 (Reference 2.2-5). The assessment considered the location of the Darlington Nuclear site, also encompassing the DNNP site, which is about 1.0 km away from the Macdonald–Cartier Freeway (also known as Highway 401) and one of the busiest highways in Canada.

The event scenario considered involves two tractor trailers crash (or rollover), such that multiple containers are damaged, consequential toxic or asphyxiant materials are released into the atmosphere, and the wind (2 m/s) disperses the airborne chemicals toward the BWRX-300 HVAC systems intakes (refer to Chapter 9A, Section 9A.5 for information on BWRX-300 HVAC systems).

Highway 401 is about 1.0 km north of the DNNP site. The impact of two tractor trailer crash is therefore screened out based on distance. Explosion or release of toxic/asphyxiant materials from the colliding two tractor trailers depends on the size of insuring breaks and the consequential amount of material released (via leaking or 100% break), wind direction and speed, and the degree of dilution due to dispersion. This scenario is therefore screened out based on distance and low impact without performing confirmatory assessment.

#### **2.2.3.4 Characterization Hazards from Marine Transportation**

The cargo vessels move along shipping lanes which are designated by the Ministry of Transport, and the nearest approach is about 27 km from the Darlington Nuclear site, per Section 3.3 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). Therefore, scenarios involving tankers or cargo ships are, in general, screened out based on distance, per screening criterion [QL3]

The consequences of a chemical leak from a tanker or a cargo ship would be mostly an environmental hazard. Depending on the exact nature, severity, and progression time of the accident as well as the consequential amount of leaked material, lake current and degree of dilution, such scenarios would have negligible impact on the quality, or the quantity of the cooling water supplied to the BWRX-300. A tanker or cargo ship accident resulting in chemical leak is screened out based on screening criterion [QL6].

The hazard of an explosion onboard a cargo ship and subsequent release of toxic gases is screened out based on screening criterion [QL3], that is, the event cannot occur at the site or close enough to the site to affect the plant.

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), a large number of small or large recreational boats or vessels travel across Lake Ontario, Winter conditions limit this traffic to about 8 months of the year. Hazards from accidents involving such recreational boats or vessels pose no significant threat to the BWRX-300 safe operation, even if the accidents occur near the lake water intake structure. St. Marys Cement Company Limited owns a pier that is about 700 m to the east of the DNNP site. Bulk carriers may load cement or unload gypsum or coal at this dock. Also, a restricted zone is established around the BWRX-300 offshore structures. The consequence of such accidents is bounded by a frazil ice hazard, and therefore, associated hazards are screened out based on criterion [QL1]

#### 2.2.4 Characterization of Stationary Non-Nuclear Accidents Hazards

The evaluation of hazards associated with stationary non-nuclear accidents is based on the results of the assessment reported in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS. Since DNNP is also located within the Darlington Nuclear site boundary, the results of the DNGS assessment are relevant to DNNP.

Event scenarios that can result in an accidental fire, explosion, or a release of hazardous material from stationary sources have been assessed in Section 3.5 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). The locations of the initiating mechanism for these sources are constrained to tank farms and forest fires.

The main stationary sources of external hazardous material near the Darlington Nuclear site are:

1. Regional Water Treatment Plants which generally have a large inventory of Chlorine for treatment of water.
2. Cigas Propane, which is located 3.6 km away from Darlington Nuclear site, where a large inventory of propane gas is stored.
3. St. Marys Cement plant located about 1.5 km east of the DNGS site and approximately 700 m from the DNNP site. The plant stores large inventories of a variety of hazardous chemicals on-site. The main toxic and hazardous materials are as follows (Reference 2.2-16):
  - Aqueous (19%) ammonia ( $\text{NH}_4\text{OH}$ ) tank with capacity of up to 38 000 L
  - Diesel fuel storage tanks with capacity of up to 50 000 L used for heating and fueling mobile equipment.
4. The DNGS Tritium Removal Facility where chemicals and fuel stored could potentially pose hazards to DNNP BWRX-300 resulting from the release of toxic chemicals, hydrocarbon explosions (Boiling Liquid Expanding Vapour Explosions and Vapour Cloud Explosions), or confined explosions (refer to Subsection 2.2.5.4 for additional information on DNGS potential hazards).

Substantial pipelines carrying large quantities of natural gas do not run close enough to the Darlington Nuclear site. Therefore, the risk of fire due to pipelines ruptures poses a negligible incremental risk to the DNNP site and, thus, it was screened out based on screening criterion [QL-6] (Subsection 2.2.2).

##### 2.2.4.1 Characterization of Toxic Chemicals Releases from Stationary Hazards

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the event scenario assessed involves a local accident in one of the regional water treatment plants (for example, Courtice WPCP) or in the St. Marys Cement plant, resulting in the release of chlorine gas ( $\text{Cl}_2$ ) or gas/aqueous ammonia ( $\text{NH}_3/\text{NH}_4\text{OH}$ ), respectively. Combustion of  $\text{NH}_3$  in the air could result in  $\text{NO}$  or  $\text{NO}_2$ , in the presence of appropriate catalysts. Nitrogen dioxide is toxic by inhalation, but it is easily detectable by smell at low concentrations. The combustion of ammonia in air is difficult in the absence of a catalyst, as the temperature of the flame is usually lower than the ignition temperature of the ammonia-air mixture.

The accident is assumed to include multiple containers. As such, the airborne toxic material, chlorine, or ammonia, released into the atmosphere could disperse toward the BWRX-300 HVAC intakes. Depending on the size and nature (i.e., severity and time frame) of the release, wind direction and wind speed, the concentration of toxic chemicals varies.

For the chlorine hazard, the nearest water treatment plant, the Courtice WPCP, is approximately 5 km west of the BWRX-300 HVAC intakes. Thus, this hazard is screened out under screening criterion [QL-3] (Subsection 2.2.2) as the event cannot occur at the site or close enough to the site to affect the plant.

With respect to the ammonia hazard associated with accidents at the St. Marys Cement plant which is located approximately 700 m east of DNNP site boundary (Reference 2.2-16) and considering the total low-level of inventory of ammonia at the St. Marys plant, the toxic release is screened out from further assessment under screening criterion [QL-1] (Subsection 2.2.2).

#### **2.2.4.2 Characterization of Explosions from Stationary Sources**

The event scenario involves the explosion of multiple propane tanks at the Cigas Propane storage facility, or the explosion of multiple diesel fuel tanks located at the St. Marys Cement plant as per Subsection 3.5.2 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). As multiple tanks are damaged, there are missiles potentially generated by the explosions, as well as shockwaves, which can damage SSCs several hundred meters away.

The screening distances for different types of explosions, per the 2019 NK38-REP-03611-10043 (Reference 2.2-5) are estimated at 1600 m for Boiling Liquid Expanding Vapour Explosion, 700 m for explosions equivalent to 61.5 Mg trinitrotoluene, and 460 m for Vapour Cloud Explosion. For the DNNP, considering the distances of the hazardous sites (3,600 m for Cigas Propane, and greater than 700 m for St. Marys Cement), both scenarios for Boiling Liquid Expanding Vapour Explosion due to propane tanks explosions at Cigas Propane, and explosions due to diesel fuel tanks at St. Marys Cement were screened out, based on distance screening criterion [QL3]. (NOTE: The St. Marys Cement does not store large quantities of pressurized light hydrocarbons (unlike that in Cigas Propane.)

An assessment of missiles generated from an explosion associated with hydrogen used in the Tritium Removal Facility was performed in 2019 for DNGS in NK38-REP-03611-10043 (Reference 2.2-5). The Tritium Removal Facility is located directly west of the DNGS vacuum building. The assessment determined that missiles generated by an explosion in the Tritium Removal Facility are bounded by missiles generated by a design basis tornado, for which DNGS is protected.

The DNNP facility is approximately 1.0 km away from the Tritium Removal Facility, and the DNGS and its vacuum building provide an obstruction between the Tritium Removal Facility and the DNNP BWRX-300 facility. As such, this hazard is screened out based on [QL-3] for large missiles since the event cannot occur on or close enough to the DNNP site to affect the BWRX-300 facility. Small missiles generated by an explosion at the Tritium Removal Facility can also be screened out for the DNNP BWRX-300 design, using screening criterion [QL-4], since such small missiles are bounded by the design basis tornado automobile missiles (refer to Subsection 2.6.6).

#### **2.2.5 Characterization of Stationary Nuclear Accidents Hazards**

Stationary nuclear accident sources within the vicinity of DNNP that pose potential hazards from nuclear accidents are:

1. Cameco's Port Hope Uranium Conversion Facility – located about 40 km east of the Darlington Nuclear site where the DNNP is located
2. PNGS – located about 25 km west of the Darlington Nuclear site where the BWRX-300 is to be built
3. DNGS – located within one kilometer west of the BWRX-300 footprint

The Cameco facility and PNGS were assessed in the 2019 Hazard Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) performed for DNGS.

#### **2.2.5.1 Evaluation of Cameco's Port Hope Uranium Conversion Facility Hazard**

Cameco's Port Hope uranium conversion facility is a nuclear substance processing facility licensed to process uranium trioxide ( $\text{UO}_3$ ) into both uranium dioxide ( $\text{UO}_2$ ) and uranium hexafluoride ( $\text{UF}_6$ ). Natural  $\text{UO}_2$  is used to manufacture CANDU fuel for nuclear power reactors in Canada, while  $\text{UF}_6$  is exported to companies in other countries for enrichment and fabrication into fuel for international nuclear power reactors. The facility is located on the north shore of Lake Ontario, approximately 40 km east of Darlington Nuclear site. The Cameco plant is a chemical processing facility with negligible radioactive releases, and therefore it is not included in the screening analysis for DNGS. Based on the DNNP proximity to DNGS, the screening results for DNGS are directly applicable to DNNP and hence screened out from further evaluation both deterministically and probabilistically.

#### **2.2.5.2 Characterization of Pickering Nuclear Generating Station Hazards**

PNGS is located on the shores of Lake Ontario, approximately 25 km west of Darlington Nuclear site. The PNGS is an eight-unit station with six operating CANDU reactors with a total output of 3100 MWe, and two units in safe storage. OPG is conducting a re-assessment, per the 2022 P-CORR-00531-23042 (Reference 2.2-13), involving a comprehensive technical examination of the potential for refurbishing Units 5, 6, 7 and 8 of PNGS. The results including recommendations of such an assessment are to be reported in 2023.

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the accidental release of radioactive materials at PNGS can be screened out for DNGS given it is a slow developing event, and there are mitigating features as well as enough time for operators to take proper actions. As the DNNP is farther from PNGS and similar mitigation measures, if warranted, are implemented, the radiological hazards associated with such events are also screened out for DNNP. Any hazard from PNGS used CANDU fuel still within an irradiated fuel bay or a dry storage facility is bounded by the much closer event from DNGS discussed in Subsection 2.2.5.4.2.

#### **2.2.5.3 Characterization of Darlington Nuclear Generating Station Hazard**

The BWRX-300 Unit 1 footprint resides partly within the DNGS exclusion zone (nominally 914 m), that is within DNGS controlled area, per Subsection 5.10 of the 2020 NK054-CORR-00531-10533 (Reference 2.2-4). The closeness of DNNP to DNGS means that in the event of a nuclear accident within DNGS the ability to maintain safe operation of DNNP can potentially be affected.

#### **2.2.5.4 Characterization of Other Radiological Hazards from DNGS**

Potential radiological hazards in the area that could affect the safe operation of the new nuclear plant were evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). Nuclear events at the DNGS considered in this assessment were as follows:

- Tritium Removal Facility accidents leading to release of tritium
- In-plant fire near a storage area of active liquid waste
- Used irradiated fuel accident
- Design basis reactor accidents
- Beyond design basis reactor accidents which include severe accidents that have the potential for a significant off-site release of radioactive materials

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

The assessment determined these events do not pose a concern to equipment but would likely impact the operating staff of the proposed plant (that is, the BWRX-300 facility). Four specific events as listed below are discussed in more details:

- Tritium Removal Facility – Subsection 2.2.5.4.1
- Irradiated Fuel Storage Facility – Subsection 2.2.5.4.2
- Used Fuel Dry Storage – Subsection 2.2.5.4.3
- Radioactive Waste Storage -Subsection 2.2.5.4.4

In October 2021, DNGS Power Reactor Operating Licence PROL 13.02/2015 was amended to authorize unit 2 to produce molybdenum-99, an isotope used in the medical industry for diagnostics. The CNSC decision concludes that the licensed activities will have a negligible effect on severe core damage frequency and large release frequency (Reference 2.2-17). In the future, DNGS may pursue production of other isotopes and/or molybdenum-99 in other units.

***2.2.5.4.1 Characterization of Tritium Removal Facility Hazard***

The Tritium Removal Facility is located within the boundary of the DNGS site, to the west side of the DNGS vacuum building. Release of tritium from an accident at the Tritium Removal Facility was evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). The assessment concluded that accidents leading to a tritium release do not pose concern to equipment but have the potential to impact operators. (Refer to Chapter 6, Subsections 6.4.1.1 and 6.4.1.2 for information on the BWRX-300 MCR and SCR habitability provisions, respectively.)

Helium-3 (He-3) is also extracted from tritium storage containers at the Tritium Removal Facility for medical and commercial uses. He-3 is a non-radioactive, inert, and non-toxic gas and therefore accidental release does not contribute any additional risk.

***2.2.5.4.2 Characterization of Irradiated Fuel Storage Facilities Hazards***

Following its useful life in the DNGS reactors, used CANDU fuel bundles are discharged from the fueling machine heads and initially stored underwater in modules in irradiated fuel bays at the West and East Fueling Facility Auxiliary Areas, located inside the DNGS protected area, adjacent to Unit 1 and Unit 4, respectively. Then the used fuel modules are transferred to and placed into seismic stacking frames inside the main irradiated fuel storage bays where the water in the bays removes heat produced by the decaying used fuel and provides shielding for workers. After a specified number of years, based on bays capacity and operational needs, the used fuel is transferred to an on-site irradiated fuel dry storage facility for short-term storage, and ultimately to an off-site long-term dry storage facility when it becomes available in the future. The hazards posed by both the irradiated fuel bays and the on-site irradiated fuel dry storage facility are analyzed in NK38-REP-03611-10043 (Reference 2.2-5).

Radiological releases from used fuel accidents were also evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). It was determined that used fuel accidents posed no concern for DBAs.

Analysis of human-induced hazards and natural hazards for the DNGS irradiated fuel bays was performed and documented in Section 5 and Section 6, respectively of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). All human-induced hazards analyzed have been screened out (Table 5-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5)), which is applicable to DNNP as well. For natural hazards, Table 6-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5) summarizes hazards which are not screened out. Irradiated fuel bay accident analysis is documented in Subsection 3.6.4 of the 2017 NK38-SR-03500-10002 (Reference 2.2-9).

#### ***2.2.5.4.3 Characterization of Used Fuel Dry Storage Hazard***

Analysis of human-induced hazards and natural hazards for irradiated CANDU fuel dry storage facility was performed and documented in Section 7 and Section 8, respectively, of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). All human-induced and natural hazards analyzed have been screened out as not having a safety impact on DNGS. The results are directly applicable to DNNP BWRX-300 and have been screened out, as per Table 5-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5).

#### ***2.2.5.4.4 Characterization of Radioactive Waste Storage Hazard***

The scenario analyzed in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) for radioactive waste storage accidents is an in-plant fire near a storage area of active liquid waste. This event poses no concern for DBAs.

### **2.2.6 Characterization Industrial Hazards**

The primary industrial hazard of concern is uncontrolled underground blasting associated with the St. Marys Cement plant.

This hazard was assessed in Section 3.6 of the 2019 DNGS Hazard Screening Assessment NK38-REP-03611-10043 (Reference 2.2-5). The results of the assessment indicated blasting at St. Marys quarry leads to shock waves in the ground travelling to the Darlington Nuclear site.

Vibration monitors on the Darlington Nuclear site at the St. Marys' property boundary are designed to record the amplitude and frequencies of such shock waves, originating from the St. Marys Cement plant. St. Marys Cement commits to not carry out blasts that may exceed the maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s measured at the Darlington Nuclear site property boundary with St. Marys.

This agreement was originally put in place to avoid turbine trips at DNGS. Since DNNP is in geographic proximity to DNGS and is closer to St. Marys Cement plant than DNGS, this hazard is applicable to the BWRX-300 facility.

The agreement noting 3 mm/s is between OPG and St. Marys and is therefore applicable to DNNP.

The maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s measured at OPG's Darlington Nuclear site property boundary is screened in and shall be considered in the design of the BWRX-300 facility.

### **2.2.7 Characterization of Biological, Animal, and Frazil Ice Hazards**

Lake Ontario is the reservoir of cooling water for the DNNP BWRX-300 facility. Fouling of the intake structures and components from growth of biological species (e.g., algae, mussels, or clams) and the presence of animals (e.g., birds, fishes, or other wildlife) impede the availability of water for heat sink purposes. Also, the formation of frazil ice at the intake can restrict or block supply to the Circulating Water System (CWS) (refer to Subsection 2.5.2). Both potential hazards are evaluated in the following two subsections.

#### **2.2.7.1 Characterization of Biological and Animal Hazard**

##### **Biological Hazards**

A variety of sources of organisms or organic material that could contribute to biofouling associated with cooling water systems originate from the pathway represented by Lake Ontario, thus restricting or blocking water supply to the BWRX-300 facility.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

The impact of biological and animal hazards on the safe operation of DNNP was considered and documented in the 2009 NK054-REP-01210-00019 (Reference 2.2-2), the 2020 NK054-CORR-00531-10533 (Reference 2.2-4), the 2009 NK054-REP-01210-00018 (Reference 2.2-7), and the 2021 NK054-REP-01210-00078 (Reference 2.2-8).

Section 2 of the 2009 NK054-REP-01210-00018 (Reference 2.2-7) assessed the hazards associated with blockage of cooling water intake. The primary species that can contribute to biofouling have been identified and assessed. Biofouling was identified as a potential hazard that can result in loss of cooling and fouling of cooling equipment, such as lines and heat exchangers.

Section 3.5 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) considered the adequacy of water supply affected by biofouling, where several species were assessed.

Further discussion on the prevention of biofouling for the cooling water intake is provided in Subsection 2.5.2.2.

#### Animal Hazards

Airborne animal hazards (e.g., birds or insects) have the ability to block the screens of the MCR air ventilation intakes. This event is equivalent to outside air damper isolation during off-normal conditions, as described in Chapter 9A, Subsection 9A.5.2.1.4. The airborne animal hazard is therefore screened out using screening criterion [QL6].

#### **2.2.7.2 Characterization of Frazil Ice Hazard**

Section 3.5 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) states frazil ice forms in turbulent, supercooled water (water temperatures of  $-0.01^{\circ}\text{C}$  to  $-0.05^{\circ}\text{C}$ ). To generate these conditions, hydro-meteorological conditions must be such that there is sufficient heat loss from the water to cause water temperature to decrease to the freezing point. The physical parameters relevant to the formation of frazil ice include water temperature, air temperature, wind speed, and humidity.

In lakes, blockages of intakes are associated with open water, low temperatures, and clear nights. They also are often associated with strong winds, which increase the rate of heat loss at the water surface as well as potentially provide turbulence that can mix the supercooled water to the depth of the intake. The intake flow can also entrain the supercooled water if it is of sufficient velocity. The depth at which a lake intake will be free from the impacts of frazil ice is also dependent on other factors, such as lake bottom topography and intake structure dimensions.

Frazil ice is considered a potential hazard for causing water intake blockage to the BWRX-300 facility.

#### **2.2.8 Characterization of Ice Storm Hazard**

The impact of ice storms on the safe operation of the reactors at the Darlington Nuclear site was considered in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) for DNNP and assessed in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS.

Section 3.2 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) considered ice storms as part of the freezing rain assessment under rare meteorological events. The major ice storm event on record for the Darlington Nuclear site occurred in January 1998, over a period of 5 days. During the storm event, 80 -100 mm of freezing rain affected areas from Kingston to Granby, Quebec. On average, Toronto Pearson Airport recorded 17.1 hours of freezing rain per year, 8.8 days per year; while Trenton airport reported 21.9 hours of freezing rain per year and 11.4 days per year, as per the 2021 NK054-REP-01210-00078 (Reference 2.2-8).

Freezing rain totals ranging from 50 mm to 75 mm have been reported on few occasions in southern Ontario; whereas 10 mm of freezing rain is to be expected occasionally and up to 20 mm of freezing rain is highly likely to occur over the time the site will be operational. Historically, freezing rain events with more than 50 mm have been observed in the same broad climatological region but are not frequent, with maximal amounts near 100 mm (refer to the 2009 NK054-REP-01210-00008 (Reference 2.2-1)).

The ice storm hazard for DNGS was assessed in the 2019 Darlington Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) and documented there in Subsection 4.5.5. The analysis reviewed OPG and CANDU Owners Group operating experience databases, as well as databases for other power plants. The review showed ice storms have not had an impact on the plants, but severe storms were seen to lead to losses of off-site power and switchyard failures in several cases. In 1998, Hydro Quebec experienced a loss of grid for several days due to an ice storm. During this ice storm, 40 mm of freezing rain was observed in Kingston, Ontario, and as much as 120 mm of freezing rain was observed in certain parts of Quebec.

For the DNNP BWRX-300, the LOPP event, an Anticipated Operational Occurrence (AOO), which is in the Pressure Increase Group and is designated as a BL-AOO event (refer to Chapter 15, Subsection 15.5.3.2.4).

### **2.2.9 Characterization of Electromagnetic Interference Hazard**

Electromagnetic interference can affect the functionality of instrumentation and control equipment and can be initiated by both on-site sources, such as high-voltage switchgear and off-site sources such as communication networks. It has the potential of disrupting electrical components and instrumentation leading to potential impairment of critical plant control signals. This hazard was assessed in the 2009 NK054-REP-01210-00008 (Reference 2.2-1), the 2009 NK054-REP-01210-00019 (Reference 2.2-2), and the 2020 NK054-CORR-00531-10533 (Reference 2.2-4) for DNNP and in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS.

Section 2.1 of the 2009 NK054-REP-01210-00019 (Reference 2.2-2) identified this hazard for consideration in the design to provide the required shielding of critical components and “fail safe” wherever required.

Section 4.9 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) assessed external sources of electromagnetic interference including high-voltage transmission lines at DNGS and telecommunications towers. The assessment concluded that since electromagnetic interference sources are continuously present (including lightning induced electromagnetic interference), the risk of electromagnetic interference at the site must be addressed in the design basis of the new plant (currently, that is the BWRX-300 facility).

### **2.2.10 Characterization of On-site Methane Hazard**

During initial site investigation, naturally occurring gas (methane) was found at/or near the bedrock/overburden interface in Boreholes DN-34, DN-41, DN-44, DN-48, DN53, DN-57, and DN-60 as described in Subsection 5.3.1 and Section 9.3 of the 2009 NK054-REP-01210-00011 Site Evaluation (Reference 2.2-6). Methane gas is harmful to the health of humans and is combustible. Methane is naturally produced at low-level from the bedrock by decaying vegetation from long ago.

Excavation near the bedrock/overburden interface will monitor for the methane gas and precautionary measures during construction will be taken per work documentation as required by the Canadian Centre for Occupational Health and Safety. For the RB excavation, the methane is expected to dissipate quicker than what was observed in the boreholes due to the significantly larger air space.



### **2.2.11 References**

- 2.2-1 NK054-REP-01210-00008 R001, 2009, "Site Evaluation for OPG New Nuclear at Darlington - Nuclear Safety Considerations," Ontario Power Generation.
- 2.2-2 NK054-REP-01210-00019 R000, 2009, "Identification of Potential Design Implications Resulting from the Darlington Site Evaluation Project," Ontario Power Generation.
- 2.2-3 NK054-REP-01210-00108 R000, 2019, "Site Preparation Nuclear Safety Licence Renewal Activity Report," Ontario Power Generation.
- 2.2-4 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation Licence (PRSL)," Ontario Power Generation.
- 2.2-5 NK38-REP-03611-10043 R003, 2019, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 2.2-6 NK054-REP-01210-00011 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.2-7 NK054-REP-01210-00018 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Additional Considerations," Ontario Power Generation.
- 2.2-8 NK054-REP-01210-00078 R007, 2021, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 2.2-9 NK38-SR-03500-10002 R005, 2017, "Darlington Nuclear 1-4 Safety Report: Part 3 – Accident Analysis," Ontario Power Generation.
- 2.2-10 NK054-REP-01210-00144 R000, 2022, "(DNNP) Hazard Analysis Methodology," Ontario Power Generation.
- 2.2-11 NK054-REP-01210-00150 R000, 2022, "Darlington New Nuclear Project Rail Transportation – Toxic Gas/Chemical Release Hazard Assessment," Ontario Power Generation.
- 2.2-12 NK054-REP-01210-00149 00149 R000, 2022, "Darlington New Nuclear Project Rail Transportation – Explosion Hazard Assessment," Ontario Power Generation.
- 2.2-13 P-CORR-00531-23042, 2022 "Pickering NGS – Request to Extend Deadline Under PROL 48.01/2028 Licence Conditions 15.1 and 15.4 Related to the End of Commercial Operation", Ontario Power Generation.
- 2.2-14 CNSC Regulatory Document REDGOC-2.4.1, "Safety Analysis - Deterministic Safety Analysis."
- 2.2-15 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.2-16 NK054-CORR-01210-1043237 R000, 2022, "St. Marys' Tank Location," Ontario Power Generation
- 2.2-17 CNSC DEC 21-H107, October 26, 2021, "Record of Decision – Application to amend Power Reactor Operating Licence PROL 13.02/2025 to Authorize Production of Molybdenum-99 at the Darlington Nuclear Generating Station."

## **2.3 Proximity of Industrial, Transportation and Other Facilities**

Information in Section 2.3 describes potential hazards associated with transportation network, industrial facilities and the DNGS which are proximate to the DNNP site.

### **2.3.1 Transportation Network**

There are multiple transportation networks within, adjacent to, and around the Darlington Nuclear site that present potential risks to the BWRX-300 facility operation.

#### **a. Canadian National Railway**

The Canadian National Railway line bisects the Darlington Nuclear site and is primarily used to transport commuters (VIA Rail) with services from Toronto to Kingston, Montreal, and Ottawa. Significant number of passengers travel this route annually and tremendous cargo is transported annually on the line, including coal, forest products (e.g., lumber), chemicals, petroleum products (e.g., asphalt), automotive parts/products, and agricultural goods (e.g., fertilizer).

Given the high frequency of both commuter and cargo traffic on this railway line, there is a potential risk of train derailment at the site. This risk is mitigated to some degree as the railway line is well buffered by berms on both sides of the railway corridor that would involve any possible derailment. In addition, VIA Rail announced in 2021 it was embarking the High Frequency Rail project that will divert a portion of the commuter rail to a separate line to relieve congestion on the current line and avoid congestion risks with cargo/freight shipments.

Additional information on hazards related to rail transportation accidents is provided in Subsection 2.2.3.2.

#### **b. Highway 401**

Highway 401, its official name Macdonald–Cartier Freeway, is a controlled-access 400-series highway stretching from Windsor in the west to the Ontario–Quebec border in the east. The highway runs along the north of the Darlington Nuclear site boundary as a six lane (three east-bound lanes and three west-bound lanes) highway.

Information on transportation risk associated with the 401 highway is described in Subsection 2.2.3.3.

#### **c. Oshawa Executive Airport**

The Oshawa Executive Airport, owned and managed by the City of Oshawa, is located northwest of the Darlington Nuclear site. It is located on an approximately 2.0 km<sup>2</sup> site with a modern terminal building and dual runways measuring approximately 1296 m and 809 m, respectively, to service different types of aircraft. The airport is required by the federal government to operate until 2047 but may close prior to 2047 (but not before 2033 at the earliest) if Pickering airport is opened. In 2018, total aircraft movement at the airport was over 78,000.

Information on risk associated with air transportation is presented in Subsection 2.2.3.1.

### **2.3.2 Industrial Facilities**

There are few industrial facilities in proximity to the east of the DNNP site and to the west of the DNGS site that could cause potential risks to the BWRX-300 operation. Details are presented on such facilities in Subsection 2.1.1, and on pertinent potential hazards in Subsection 2.2.4 and Subsection 2.2.6.

### **2.3.3     Darlington Nuclear Generating Station Site**

There are numerous activities at the DNGS that may impact the operation of the BWRX-300. The following activities apply:

- a. OPG uses arial photography drones, for inspection of the exterior of some of the DNGS buildings, as well as systems and components. The hazard of such drone crashing on the BWRX-300 buildings is bounded by the design basis automobile tornado missiles (refer to Subsection 2.6.6).
- b. Chemicals and gases used at the 2019 DNGS NK38-REP-03611-10043 (Reference 2.3-1) are screened out on the basis:
  - That their impact is bound by the impact of similar chemicals on the BWRX-300 (refer to Section 2.4, Table 2.4-1)
  - Of distance from the DNNP site
  - Of the probability of occurrence of relevant accidents.

Refer to Subsection 2.2.5.3 and Subsection 2.2.5.4 for additional and detailed coverage of other hazards related to the operation of the DNGS, or activities being undertaken at the DNGS site.

### **2.3.4     References**

- 2.3-1     NK38-REP-03611-10043 R003, 2019, "Hazard Screening Analysis - Darlington," Ontario Power Generation.

## **2.4 Plant Site Activities Influencing Plant Safety**

Section 2.4 includes two subsections:

- Subsection 2.4.1, which evaluates processes and activities at the DNNP site that, if incorrectly carried out, could affect or influence the safe operation of the BWRX-300 facility
- Subsection 2.4.2, which discusses measures for site and shoreline protection.

### **2.4.1 Site Hazards**

Subsection 2.4.1 is limited to processes and activities at the DNNP site. Activities at DNGS or other off-site industrial locations are considered in Section 2.3. Subsection 2.4.1 information is focused on the following site-specific sources of hazards:

- Potentially explosive gases – Subsection 2.4.1.1
- Flammable vapour clouds – Subsection 2.4.1.2
- Toxic chemicals – Subsection 2.4.1.3
- Fire and smoke – Subsection 2.4.1.4

Table 2.4-1 provides a listing of gases and chemicals stored on the DNNP site.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.4-1: Summary of Gases and Chemicals Stored on DNNP Site**

<b>Chemical/Material (Formula/Trade/State)</b>	<b>Location (subject to change)</b>	<b>Quantity</b>	<b>Hazard Screening</b>
Nitrogen	Gas Storage Area West of TB	Approximately 50 m <sup>3</sup> (Cryogenic Storage Tank)	Nitrogen is evaluated as potential asphyxiant concern for MCR and SCR habitability.
Hydrogen	Gas Storage Area West of TB	Each cylinder stores 356.1 standard cubic meters (SCM).	Hydrogen is a potential explosive and fire concern. Minimum separation distance between the cylinders and the BWRX-300 RB wall is determined based on explosive potential.
Diesel Fuel	Tank North of the Protected Area Access Building	Approximately 114 000 L Tank	Not a toxic or explosive hazard. Potential of fire hazard is addressed in Chapter 9A, Section 9A.6.
Turbine Oil	Tank North of the Protected Area Access Building	Approximately 20 000 L tank (volume of the tank does not impact MCR habitability)	Not a toxic or explosive hazard. Potential fire hazard is addressed in Chapter 9A, Section 9A.6.
Sodium Hypochlorite (7 to 15% Solution)	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 4000 L tank	Sodium hypochlorite is not considered a hazard due to being a liquid at 37.8 °C (100 °F) and normal atmospheric pressure. Sodium hypochlorite has a relatively low vapour pressure. Due to the relatively low vapour pressure, no significant unreported and prolonged release that could affect MCR habitability would be expected even in the event of a major spill.
Sodium Bisulphite (24 – 38% Solution)	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 11 400 L tank	Based on chemical safety data sheet sodium bisulphite is relatively stable. Sodium bisulphite is not considered a hazardous substance based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health.
Captor Thiosulphate Dichlorination	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 11 400 L tank	Based on chemical safety data sheet captor thiosulphate is not a toxic hazardous substance.
Gasoline	Vehicle Maintenance Garage	Approximately 20 L containers	Gasoline is a potential explosion and fire concern. Small quantities do not pose a significant hazard.
Propylene Glycol	Within the P25, Chilled Water System, throughout the Power Block	39 000 L	Based on chemical safety data sheet propylene glycol is not a toxic hazardous substance for MCR habitability.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Chemical/Material (Formula/Trade/State)	Location (subject to change)	Quantity	Hazard Screening
Tetrafluoroethane (R-134a Refrigerant)	P25 Chillers on RadWaste Building Roof	Each Chiller contains a refrigerant charge of 250 kg	R134a is not a toxic hazard for MCR habitability. Release of the entire contents of the R-134a into the Control Building does not result in an oxygen-deficient environment in the MCR.
Noble Metal Solution	Reactor Building	Approximately 38 L of 1% noble metal solution is utilized over a 2-week time frame per year.	The noble metal solution is not considered a hazard to MCR habitability based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health. A potential release will be relatively confined to the RB and not impact MCR habitability.
Depleted Zinc Oxide	Turbine Building (TB)	90 kg dissolution vessel (quantity does not impact MCR habitability)	Zinc oxide is not considered a hazard to MCR habitability based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health. A potential release of zinc oxide dust will be relatively confined to the Turbine Building and not impact MCR habitability.

#### **2.4.1.1 Potentially Explosive Gases**

The nearest source of potentially explosive gases is the hydrogen gas storage cylinders for the Reactor Water Chemistry System. Table 2.4-1 lists the maximum quantity of hydrogen stored at this location. The hydrogen is stored in several cylinders.

The safe separation distance between the hydrogen storage area and nearest safety-related structure is determined using a methodology such as the approach in EPRI NP-5283-SR, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations, September 1987 (Reference 2.4-1). In the 1987 EPRI NP-5283-SR (Reference 2.4-1) the required separation distance is determined for two different considerations. The first consideration is the required separation distance such that the safety-related structure is not adversely affected by the postulated hydrogen explosion. The second consideration is the required separation distance to air pathways into safety-related structures versus the internal diameter of leaking high-pressure piping. The results of the determination of required separation distance are considered in establishing the layouts for the DNNP site and BWRX-300 facility.

#### **2.4.1.2 Flammable Vapour Clouds**

There are no liquids stored on the DNNP site that can generate a significant quantity of flammable vapour.

#### **2.4.1.3 Toxic Chemicals**

Table 2.4-1 identifies the chemicals on the DNNP site that are considered in the evaluation of potential toxic chemical hazards. Table 2.4-1 identifies the chemical, the quantity, and how the chemical is dispositioned. Chemicals are initially evaluated based on relative location, quantity stored, toxicity, and properties such as vapour pressure. As shown in Table 2.4-1, from a toxic chemical perspective, the potential hazards at the DNNP site except for nitrogen are dispositioned as not being hazardous for control rooms habitability. The liquid nitrogen, however, cannot be screened out and requires a detailed evaluation.

The threat from nitrogen is displacement of oxygen. No specific acceptance criterion is provided for limiting concentrations, and nitrogen is not considered a toxicity hazard. Nitrogen impacts control room habitability if it displaces sufficient quantities of air to the extent that oxygen levels in the room decrease below a specified threshold. Chemicals are asphyxiating if they result in an oxygen-deficient atmosphere of less than 19.5% oxygen by volume, as defined by the Canadian Centre for Occupational Health and Safety.

As described in Chapter 6, Section 6.4, control room habitability is served by a combination of individual systems that collectively ensure that continued occupancy in the MCR or SCR is possible under Postulated Initiating Events (PIEs) for a minimum of 72 hours as required by REGDOC-2.5.2 (Reference 2.4-4).

Two different scenarios are considered: a tank burst and a tank leak. In the tank burst scenario, all the contents of the tank are instantaneously released. For the tank leak scenario, the nitrogen is leaked at a constant mass flow rate until the tank is empty over an assumed time. Inputs to the analyses include meteorological stability classification, wind speed, air temperature, and the assumed leak rate for the tank leak scenario. Several sensitivity cases are run to determine the limiting input values. For each location, the control room ventilation system is modeled in the analyses to credit the effects of intake and dilution within the control room atmosphere during the passage of the plume.

The limiting results from the analyses of the postulated nitrogen tank burst and leak scenarios are used to confirm that the placement of the tank relative to the MCR and SCR ventilation intakes is acceptable.

#### **2.4.1.4 Fire and Smoke**

On-site flammable and combustible liquid or gas storage facilities are designed in accordance with applicable fire codes, and plant safety is not jeopardized by fires or smoke in these areas. A detailed description of the fire protection system, as well as the Fire Hazard Assessment (FHA) methodology is presented in Chapter 9A, Section 9A.6.

#### **2.4.2 Measures of Site Protection**

As described in Subsection 2.7.1, the plant grade elevation at 88 m CGD is established using grading and engineered fill. Excavation is performed to depths to reach materials of specific properties suitable for buildings foundations. Materials removed during the excavation are reconditioned for use as backfill material if the material meets the required specifications or are disposed as spoils. Engineered fill material requirements are specified in Subsection 2.7.5.2.1.

The hydrology for the site and vicinity is described in Section 2.5. The site does not credit dams or dikes for flood protection. As described in Section 2.5 the topography and grading at the plant site are considered in the site flooding analyses to demonstrate the plant is adequately protected from precipitation events.

As described in the 2022 NK054-REP-07730-00055 DNNP Environmental Impact Statement [EIS] Review Report for BWRX-300 (Reference 2.4-2), the BWRX-300 deployment will not require lake infilling and, consequently, the associated adverse effects on site drainage and water quality will not occur. The BWRX-300 deployment will still require some shoreline protection works, but such works are expected to be smaller in scale resulting in smaller residual adverse effects on shoreline processes than those assessed in the 2009 EIS for no specific reactor technology NK054-REP-07730-00029 (Reference 2.4-3).

The construction of the first BWRX-300 would provide an opportunity to retain the Bank Swallow nesting habitat as the bluff would be remaining in place and as the impact of excavation and construction activities will be kept to a minimum, per the 2022 NK054-REP-07730-00055 (Reference 2.4-2). When the DNNP site is built out to include a total of four BWRX-300 reactors, additional shoreline protection is to be implemented to stabilize the shoreline as described in the 2022 NK054-REP-07730-00055 (Reference 2.4-2).

The specific extent and location of the shoreline protection works is determined in later phases of the project.

#### **2.4.3 References**

- 2.4-1 EPRI NP-5283-SR, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987, Electric Power Research Institute.
- 2.4-2 NK054-REP-07730-00055, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.4-3 NK054-REP-07730-00029, 2009, "Environmental Impact Statement New Nuclear - Darlington Environmental Assessment," Ontario Power Generation.
- 2.4-4 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."



## **2.5 Hydrology**

### **2.5.1 Introduction**

Section 2.5 describes the hydrological conditions and their potential implications relevant to the DNNP site. Section 2.5 includes information on:

- The adequacy of the cooling water supply from Lake Ontario along with risks to the water supply (i.e., biofouling and frazil ice) - Subsection 2.5.2
- The potential flooding hazards, including the Probable Maximum Precipitation (PMP), Probable Maximum Flood (PMF), as well as flooding potential from runoffs, rivers, waves, storm surge and seiche, tsunami, and ice jamming - Subsection 2.5.3
- The potential impact of climate change - Subsection 2.5.4
- Assessment and monitoring of radionuclide dispersion in the groundwater – Subsection 2.5.5
- Assessment and monitoring of radionuclide dispersion in surface water – Subsection 2.5.6

Key hydrological characteristics and parameters described in Section 2.5 relevant to the DNNP site and the surrounding area are summarized and listed in Table 2.5-1. The list includes information on Lake Ontario adequacy as a water supply for use as a heat sink, maximum precipitation and flooding and associated probabilities, as well as surface and subsurface geotechnical properties relevant to transport of radionuclides.

**Table 2.5-1: Hydrological Characteristics Summary of DNNP Site and Surrounding Area**

2.5.2 Description of Heat Removal Methods and Heat Sink			
Normal Heat Removal / Normal Heat Sink	The NHS is a once-through cooling water source from Lake Ontario to the CWS and the PCW		
Ultimate Heat Removal / Ultimate Heat Sink	The Isolation Condenser System consists of three high-pressure reactor isolation loops that passively remove heat from the reactor when the normal heat removal system is unavailable.		
2.5.2.1 Description of Lake Ontario Water Levels and Adequacy of Water Supply			
Water Level	Controlled by the International Joint Commission		
Variability of Water Level (at the intake)	Lowest water level	73.71 m (statistical data at Cobourg Water Level Station) (Reference 2.5-18)	→ 73.71 m
	Impact of seiche	0.75 m (reduction)	→ 72.96 m
	Wave trough (1 s passage)	4.08 m (reduction)	→ 68.88 m
	Spring tides	Less than 5 cm (hidden as part of normal fluctuation)	→ 68.88 m
	Wave downwash	Close to the shoreline with no effect	→ 68.88 m
	Tsunami	No risk expected	→ 68.88 m
Water Depth Available	Normal Conditions 73.71 – 62.50 m	11.21 m above the intake level of 62.50	Therefore, water supply is adequate under normal and extreme conditions
	Impact of Seiche 72.96 – 62.50 m	10.46 m above the intake level of 62.50	
	Impact of Wave Trough (1s duration) 68.88 – 62.50 m	6.38 m above the intake level of 62.50 m	
2.5.2.2 Potential Impacts of Biofouling on Water Supply			
Algae	Algae have the potential to be entrained at cooling water and water supply system intakes, resulting in blockage or restriction issues.		
Micro-biologicals	Biological coatings or biofilms and particulate deposition on tube surfaces can cause lost flow capacity, extensive repairs and material replacement costs in heat exchangers, fire protection systems, storage vessels, intakes, and water distributions systems.		
Macrophytes	Macrophytes can contribute to macrofouling through sticks, leaves and other plant constituents from either terrestrial or aquatic sources that become a component of lake drift and debris material.		
Mollusks	Zebra and quagga mussels can clog water intake structures, such as screens, tunnels and pipes.		
Fish	Various life stages of fish can be taken into a cooling water system with the cooling water (entrainment), and consequently fish reach screens that protect the cooling water and other water systems (impingement). An excessive load of fish can cause blockage to the screening system and sump. In extreme events where screens become overloaded water supply can be reduced with associated reduction in power supply.		
2.5.2.3 Potential Impacts of Frazil Ice Accumulation on Water Supply			
Frazil Ice Accumulation	Accumulation of frazil ice on the intake trash rack, which can partially or completely block the trash rack and rapidly and unexpectedly shut down the intake facility		

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.5.3 Description of Potential Sources of Flooding		
2.5.3.1 Flooding Due to PMF	PMP	420 mm 12-hour precipitation, equivalent to 420 mm of total rainfall, with 51% in the 6 <sup>th</sup> hour with a return period of 1:1,000,000 year (Reference 2.5-19)
	Design Basis Flood	<u>Conservative Rainfall</u> : Standardized value of 12-hour PMP in Ontario of 420 mm, with zero infiltration (which greatly exceeds Hurricane Hazel in depth and intensity)
	PMF – Screened in	The event scenario involves a large volume of water runoff flooding the site (based on the application of PMP), while the drainage systems are blocked (due to debris or ice pellet), the soil nearby is saturated, and the lake water level is at 100-year high. Also, it is conservatively assumed that there is no time for implementing preventative measures or taking mitigating actions. The PMF sequence is expected to be worse than a lake level increase or heavy precipitation alone, and the event is not bounded by any other events. As such, flooding due to PMF could not be screened out based on screening criteria [QL1] through [QL5].
	Design Basis Flood Level	Using design basis flood (that is, PMP with zero infiltration), for modeling drainage for BWRX-300 Unit-1 or an assumed 4-unit layouts, Section 5.4.3 of Reference 2.5-18 resulted in flood water levels of up to 87.93 m CGD, considering realistic assumptions for stormwater infrastructure, including factors such as culverts sizing, conveyance, routing, and ponds.
2.5.3.2 Flooding Due to Runoffs	Natural or via Stormwater Management and infrastructure	<ul style="list-style-type: none"> <li>• Five of nine catchments drain directly to Lake Ontario or to Darlington Creek watershed.</li> <li>• Remaining four catchments close to the BWRX-300 development area drain through a stormwater infrastructure directly to Lake Ontario and via engineered culverts stormwater infrastructure running to the southeast of DWMF to Lake Ontario (Reference 2.5-18).</li> <li>• Measures are proposed to mitigate the impact of PMP flooding due to runoff.</li> </ul>
	Screened out, per [QL2]	PMF bounds flooding caused by runoffs.
2.5.3.3 Flooding Due to Rivers	Screened out, per [QL3]	The distance, infrastructure, and topography between the Tooley Creek watercourse and the DNNP site precludes Tooley Creek as the source of a flood hazard. There is not any history of severe flooding along Darlington Creek within the recorded history of the area. This is confirmed by the (2022) Flood Hazard Assessment (Reference 2.5-18) that modeled drainage of Darlington Creek watershed under 100-year recurrence PMP.
2.5.3.4 Flooding Due to Waves	Screened in (related to Shoreline Protection issue)	Wave height of 6.1 m and peak period of 9.2 s is recommended (Reference 2.5-18) Data and models suggest wave uprush between 3.5 to 11.3 m, and overlapping from 0.015 to 0.591 m <sup>3</sup> /s/m. (Reference 2.5-2)
2.5.3.5 Flooding Due to Storm Surge and Seiche	Screened out	Models of most severe weather systems predicted a highest water level from storm surge or seiche of 0.75 m, per (Reference 2.5-2), and (Reference 2.5-18). The margins between the lake level and the top of the DNNP breakwater works are larger than 0.75 m.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2.5.3.6 Flooding Due to Tsunami	Screened out, per [QL3]	A tsunami in Lake Ontario is an improbable event for DNNP.
2.5.3.7 Flooding Due to Ponds, Dams or Dikes	Screened out, per [QL3]	There are no large permanent human-made water storage ponds, dams or dikes near the Darlington Nuclear site that can threaten the site.
2.5.3.8 Flooding Due to Ice Jamming	Screened out per [QL2] or [QL3]	Bounded by the detailed PMF analysis (Reference 2.5-4); or based on the conclusion of negligible ice forming in Lake Ontario near the DNNP region (Reference 2.5-18).
2.5.4 Potential Effects of Climate Change		
Effect on Temperature, Precipitation. Lake Water Level	Screened in	<ul style="list-style-type: none"><li>Some models showed increase in the intensity (about 14%) and frequency (about 22%) of extreme precipitation in southern Ontario (Reference 2.5-2)</li><li>Maximum found historical lake water level is 75.6 m, which should be used as low estimate (Reference 2.5-13)</li><li>For additional information, refer to the 2022 "Flood Hazard Assessment NK054-02730-00001 (Reference 2.5-18) and the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.5-20)</li></ul>
2.5.5 Groundwater		
2.5.5.1 Groundwater Conditions	Described in detail in Subsection 2.7.3.2.4.	
Groundwater Flow System	Categorized into three hydrostratigraphic units: Shallow/Water Table; Interglacial Deposits; and Shallow Bedrock. In general, groundwater flows from north to south, and discharges toward Lake Ontario.	
Groundwater Level	Groundwater is anticipated to be present between elevation 80 to 85 m corresponding to depths of between 3 and 8 m below the plant grade at elevation 88 m.	
Monitoring	Environment Monitoring Program is employed along with the use of groundwater wells that are located in key areas of the Darlington Nuclear site, including protected areas, controlled areas, and site perimeter.	
2.5.6 Surface Water		
2.5.6.1 Surface Water Properties		
Water movement near the site is predominantly along the shore, occurring for 73% of the time (35% to the west and 38% to the east).		
Depth Averaged Speed – all directions	12.4 cm/s	
Depth Averaged Speed – Easterly	14.1 cm/s	
Depth Averaged Speed – Westerly	11.3 cm/s	
Temperature	Lake-wide surface temperatures typically range from freezing in winter to approximately 20 °C in summer.	
Ice Conditions	Typically, are limited to the nearshore areas at the eastern end of the lake within the Kingston Basin.	
2.5.6.2 Surface Water Monitoring		
Lake Current Monitoring	A real-time current profile measurement system to be used in the event of a radiological liquid emission.	
Monitoring	Environment Monitoring Program is employed along with the Lake Current Monitoring system which a real-time current profile measurement system to be used in the event of a radiological liquid emission.	

## **2.5.2 Description of Heat Removal Methods and Heat Sink**

The NHS System that is described in Chapter 9A, Subsections 9A.2.5 provides cooling water source and heat rejection means to support the function of the Circulating Water System (CWS) (Chapter 10, Section 10.8) to supply cooling water to the MCA system (Chapter 10, Section 10.5), as well as to interface with the PCW (Chapter 9A, Subsection 9A.2.1). The NHS is a once-through cooling system using water from Lake Ontario. The water flows through the intake tunnel via the onshore intake vertical shaft to the Pumphouse/Forebay where the circulating water pumps deliver the cooling water to the MCA and PCW heat exchangers before returning the heated water back to the lake via the onshore discharge vehicle shaft through the discharge tunnel to the risers/diffusers.

The BWRX-300 Isolation Condenser System (ICS), described in Chapter 6, Section 6.2, consists of three independent trains, each containing a heat exchanger or Isolation Condenser (IC) that is submerged in a dedicated pool of water. The ICS provides the ultimate heat sink for protecting the reactor core for any off-normal event where the main condenser is not available, and the Reactor Pressure Vessel (RPV) is isolated.

The ICS Pool Cooling and Cleanup System (ICC) that is described in Chapter 9A, Subsection 9A.2.6 is designed to precondition and maintain the ICS pools in a state of readiness for postulated events that require reactor decay heat removal.

The FPC, as described in Chapter 9A, Subsection 9A.1.3, has a primary function to provide continuous cooling of the water volume in the fuel pool to remove decay energy from irradiated fuel, and to provide replacement coolant inventory from a variety of sources, both to ensure irradiated fuel is kept cool and submerged under water throughout the life of the plant.

### **2.5.2.1 Description of Lake Ontario Water Levels and Adequacy of Water Supply**

Lake Ontario is one of the main reservoirs of cooling water for the DNNP site. An assessment for the adequacy of water supply to DNNP was completed in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) and validated in the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18), as described in the following paragraphs.

The water level in Lake Ontario is regulated by the International Joint Commission to reduce damages along the shores of the lake and the St. Lawrence River, per the 2022 Flood Hazard Assessment NK054-REP-02730-00001(Reference 2.5-18). The control of water levels by the International Joint Commission continues in the future and, though the plan for regulation may change, the fundamental function of eliminating extreme lake levels remains. However, the International Joint Commission acknowledges that it may become increasingly difficult to maintain levels within their currently defined operating band depending on the relevant impact of climate change in the future (refer to Subsection 2.5.4 which discusses the impact of climate change on Lake Ontario water levels). Careful consideration of the International Joint Commission study for management options, which included robust modeling of potential future levels under a range of stochastically generated hydrological and meteorological conditions, led to estimates greater than 100-year recurrence low water levels at 73 m as reported in Subsection 5.1.5 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2). However, analysis of historical data at the Water Survey of Canada Cobourg Water Level Station shows a minimum water level of 73.71 m, as reported in the 2022 Flood Hazard Assessment, NK054-REP-02730-00001 (Reference 2.5-18).

Additional factors which influence the minimum water level at the intake were considered in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) as follows:

1. A numerical model of the hydrodynamics of Lake Ontario was developed to assess the potential for generation of surge and seiche in response to extreme severe weather

systems tracking through the region. The maximum wave heights expected at the intake location will be depth limited. The lowest water level of 73.71 m, further lessened by 0.75 m due to seiche, yields an elevation of 72.96 m or a depth of 10.46 m at the intake of an elevation of 62.50 m.

2. Estimating wave breaking at about 0.78 times the water depth of 10.46 m yields maximum wave heights of about 8.08 m. An associated wave trough, taken as half the maximum wave height (that is 4.8 m), might reduce the depth to 6.38 m, though it is noted that the passage of large waves would be short-lived and on the order of 1s. (Note: The 8.16 m Maximum wave height is more conservative than the maximum wave height of 6.1 m recommended in Subsection 2.5.3.4.)
3. The largest spring tides in Lake Ontario are less than 5 cm in height and these minor variations are hidden by greater fluctuations in lake levels produced by wind and barometric pressure changes. Consequently, Lake Ontario is considered to be essentially non-tidal.
4. Wave downrush would occur within a relatively close distance to the shoreline and would have no effect on the water level near the intake.
5. The 2009 flood hazard assessment (Reference 2.5-2) concluded there is no risk of tsunamis so that there is no drawdown potential from that phenomenon that could affect nearshore lake levels. The 2022 Flood Hazard Assessment (Reference 2.5-18) also concluded the Darlington Nuclear site lies in a region with a low probability of tsunamis.

Consequently, even under the extreme scenario considered in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) and the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18), a depth of more than 6 m remains over the intake at the lakebed elevation. Therefore, lake water supply is adequate for the DNNP cooling water intake.

Given the adequacy of the water supply from Lake Ontario, the potential for using groundwater sources in extraordinary situations is not considered.

Consideration for additional factors which might impact the availability of the cooling water supply were also assessed in the 2009 NK054-REP-01210-00018 (Reference 2.5-1), namely concerns related to biofouling and frazil ice conditions. These are discussed separately in the following two subsections.

Additional information on Lake Ontario's current, temperature, and ice conditions is provided in Subsection 2.5.4.2.

#### **2.5.2.2 Potential Impacts of Biofouling on Water Supply**

1. Algae: The Lake Ontario shoreline provides a favorable growth environment for *Cladophora* which are prominent nuisance filamentous algae that have the potential to affect the DNNP. *Cladophora* characteristically grows attached to hard surfaces within the littoral zone and where habitat conditions are optimal, thick mats of the algae can form across the lake substrates and become attached to infrastructure features. During mid-summer and fall, *Cladophora* senesces, the algae become detached from the substrate and drift in a suspended manner with waves and currents.

The loose filaments as well as more substantial clumps of algae have the potential to be entrained at cooling water and water supply system intakes, resulting in blockage or restriction issues at the inlet as well as further blockage and organic material loading at the trash racks or travelling screen system.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2. Micro-biologicals: Biofilms consist of microorganisms immobilized at a substratum surface, typically embedded in an organic polymer matrix of bacterial origin. Such biofilms are ubiquitous in flowing aqueous environments, are not necessarily uniform in time and space, and may trap inorganic substances within the polymer matrix. Biofilms develop on virtually all surfaces immersed in natural aqueous environments, irrespective of whether the surface is biological (aquatic plants and animals) or abiological (stones, particles, metal, and concrete, etc.). Extensive bacterial growth, accompanied by excretion of copious amounts of extracellular polymers, thus leads to the formation of visible slimy layers (biofilms) on solid surfaces.

Thin biological coatings or biofilms associated with microorganisms can reduce the efficiency of heat exchangers (forcing shutdowns or de-rating), enhance silt and particulate deposition on tube surfaces (causing fouling and pipe wall pitting), lost flow capacity, extensive repairs and material replacement costs in heat exchangers, fire protection systems, storage vessels, intakes, and water distributions systems.

3. Macrophytes: Both terrestrial and aquatic plants can contribute to floating and suspended plant material that becomes susceptible to entrainment at water intakes. A variety of rooted aquatic macrophytes are common to Lake Ontario. The existing DNGS forebay was shown to contain a community of Eurasian watermilfoil (*Myriophyllum spicatum* L.), the only rooted plant observed. The biomass of this material was estimated at 1.5 tons providing an indication of the potential availability of organic mass that can contribute to the load on the screening system. A future regional increase in aquatic plants and algae was concluded as being a reasonable expectation as the lake water clarity increases with the filtering effects of the exotic invader zebra and quagga mussel.

Macrophytes can contribute to macrofouling through sticks, leaves and other plant constituents from either terrestrial or aquatic sources that become a component of lake drift and debris material. During the fall season when macrophytes typically senesce, the organic material of the plant stems and foliage have the potential to fragment and block travelling screens.

4. Mollusks: Lake Ontario contains confirmed populations of non-native invasive nuisance mussels including the zebra mussel, *Dreissena polymorpha*, and the quagga mussel, *Dreissena rostriformis bugensis*, inadvertently introduced to North America in the ballast water of oceangoing ships. More recent colonization has involved the quagga mussel, which has a preference for deeper, cooler water as compared to the zebra mussel and has now largely replaced the zebra mussel. Given the record of non-native introductions to Lake Ontario, additional nuisance mollusk species may appear in the future. The Asiatic clam *Corbicula fluminea* has been recorded in North America the longest of the three key invasive species arriving on the west coast in the 1920s and reaching the east coast by 1980s; however, it has not yet been reported as an issue in Lake Ontario.

*Dreissena* species ability to rapidly colonize hard surfaces causes serious economic problems and potential reduced efficiency of water supply systems. These major biofouling organisms can clog water intake structures, such as pipes and screens, therefore reducing pumping capabilities for power and water treatment plants. Power plant features that may become fouled include crib structures, trash bars, screenhouses, steam condensers, heat exchangers, penstocks, service water systems and water level gauges.

5. Fish: Lake Ontario hosts a diverse population of both warm and cold-water fish species, many of which may utilize the project area either as local residents or seasonal migrants. During impingement investigations at DNGS operations from 1993 to 1995, fish encountered at the mitigative screen system and in sumps included at least 17 species. The predominant species were generally of a smaller body size which included alewife, shiner species and smelt, all representatives of the abundant forage fish-based community of the lake. Major community changes occurred with the introduction of non-native species through opening of waterways, intentional stocking, and unintentional introduction through ballast water of international shipping. This may have a bearing on future operational management systems at DNNP depending on the habits and productivity of a particular species.

Various life stages of fish can be taken into a cooling water system with the cooling water (entrainment), and consequently fish reach screens that protect the cooling water and other water systems (impingement). An excessive load of fish can cause blockage to the screening system and sumps contributing to maintenance requirements. In extreme events where screens become overloaded water supply can be reduced with associated reduction in power supply.

*NOTE: The 2009 report NK054-REP-01210-00018 (Reference 2.5-1) concludes that mitigation measures have been successfully applied at power generating facilities along the north shore of Lake Ontario to address the various forms of biofouling.*

#### **2.5.2.3 Potential Impacts of Frazil Ice Accumulation on Water Supply**

As described in the 2009 NK054-REP-01210-00018 (Reference 2.5-1), operating water intakes in lakes and rivers in northern regions is complicated by the presence of ice. Controlling the generation and accumulation of frazil ice affects both navigation and power generation. The cooling water intake tunnel can accumulate frazil ice on the intake trash rack, which can partially or completely block the trash rack and rapidly and unexpectedly shut down the intake facility.

#### **2.5.3 Description of Potential Sources of Flooding**

Subsection 2.5.3 describes the assessment of potential flood hazards at the DNNP site. (Refer to Subsection 2.1.1 for information on the topography of the Darlington Nuclear and DNNP sites.)

The review of the flood hazard assessment performed in support of the 2020 DNNP Power Reactor Site Preparation Licence (Reference 2.5-3) against the 2019 codes and standards concluded there is no impact on the conclusion of the 2009 Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.5-2) as documented in the 2019 DNNP Site Preparation Licence Renewal activity report NK054-REP-01210-00108 (Reference 2.5-7).

Also, as stated in Subsection 4.5.3 of the 2020 NK054-CORR-00531-10533 (Reference 2.5-3), the results of the 2019 Darlington Hazard Screening Analysis NK054-REP-03611-10043 (Reference 2.5-4) apply to the DNNP site since the DNNP site is encompassed in the Darlington Nuclear site.

As described and assessed in the 2019 NK054-REP-03611-10043 (Reference 2.5-4), and in the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18), the DNNP flooding hazards are:

- Flooding due to PMF - Subsection 2.5.3.1
- Flooding due to Runoffs - Subsection 2.5.3.2
- Flooding due to Rivers - Subsection 2.5.3.3



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- Flooding due to Waves - Subsection 2.5.3.4
- Flooding due to Seiche - Subsection 2.5.3.5
- Flooding due to Tsunami - Subsection 2.5.3.6
- Flooding Due to Ponds, Dams or Dikes - Subsection 2.5.3.7
- Flooding due to Ice Jamming - Subsection 2.5.3.8

These hazards are addressed in the following subsections.

#### **2.5.3.1 Flooding Due to Probable Maximum Flood**

As described in Section 5.4 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2), the design storm event used to determine the flood hazard is the PMF event in the 2011 International Atomic Energy Agency (IAEA) SSG-18 (Reference 2.5-10). This is a specific hydrologic term that is defined in conjunction with the PMP, as per the following paragraphs.

The PMF is the flood that may be expected from the most severe combination of critical meteorological and hydrologic conditions that are reasonably possible in a particular drainage area. The PMP is defined as the greatest depth of precipitation for a given duration meteorologically possible for a given size storm area at a particular location at a particular time of year, with no allowance made for long-term climatic trends. It is a common practice that the PMF is the flood which is a direct result of the PMP. The PMP is applied to sub-basin delineations that account for variations in soil type, land use, size and shape of the watershed, and average watershed slope to generate PMF flows.

There are two considerations when determining the PMP for a given application, the site location, and the duration of the storm event. Based on the 2017 Lakes and Rivers Improvement Act Technical Guidelines (Reference 2.5-11), for watershed areas less than 1295 km<sup>2</sup>, the PMP maximum precipitation duration of 6 or 12 hours is normally used as it produces the highest peak flood flow

Subsection 4.4.1 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4) states that the Review Level Condition assumes no runoff in the worst hour of the 12-hour PMP; therefore, the flood depth is 51% of the total 12-hour PMP of 420 mm, which is approximately 214 mm, per Table 5.4-1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18). The PMF event scenario involves a large volume of water runoff flooding the site, while the sewer systems are blocked (due to debris or ice pellet), the soil nearby is saturated, and the lake level is at 100-year high. This PMF sequence is expected to be worse than a lake level increase or heavy precipitation alone, and the event is not bounded by any other events. Finally, it is conservatively assumed that there is no time for implementing preventative measures or taking mitigating actions. As such, flooding due to PMF could not be screened out based on screening criteria [QL1] through [QL5] (refer to Subsection 2.2.2 for descriptions of the screening criteria).

The PMF values which are commonly estimated using a combination of flood-inducing drivers such as snowmelt and rainfall can alternatively be estimated using an extreme rainfall outside the snow season that is higher than spring values. In Subsection 5.4.1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18), it is assumed that the summer PMP produces extreme floods (i.e., PMFs) at least comparable to the spring PMFs that consider snowmelt. This assumption was verified by comparing the precipitation values of spring (March-April) with summer-fall (May-November); so that a summer PMP can be deemed as the key driver of the PMF, per the 2022 NK054-REP-02730-00001 (Reference 2.5-18).

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

As described in Subsection 2.1.1 of the 2022 NK054-REP-02730-00002, PMP Validation (Reference 2.5-19), the PMP for watershed areas in the vicinity of and the DNNP site is a 12-hour precipitation equivalent to 420 mm of total rainfall, with 51% of the storm falling in the sixth hour, with a return period of 1:1,000,000. This value is on the conservative side considering the historical observed 24-hour point rainfall in the region is 212 mm (hurricane Hazel).

The design basis flood is the flood resulting from the PMP assuming zero infiltration in the drainage areas on site. In Subsection 5.4.1 of the 2022 Flood Hazard Assessment NK054-REP-01730-001 (Reference 2.5-18) design flood values in the DNNP site region are based on the 1:100-year return period storm or Hurricane Hazel, whichever is the greater. The 1:100-year return period storm was used to calibrate a Darlington Creek model, and as a comparison to the PMP results (refer to Table 2.5-2 under Subsection 2.5.3.3). For small watersheds such as Darlington Nuclear site, where no stream gauge is available, 1:100-year return period rainfall is assumed to produce a 1:100-year return period flood. Since the 420 mm, 12-hour duration PMP greatly exceeds Hurricane Hazel in depth and intensity, Hurricane Hazel was not used in this assessment. the 420 mm, 12-hour duration PMP was selected with zero infiltration as the current design basis storm for the DNNP, as shown in Table 2.5-2.

### **2.5.3.2 Flooding Due to Runoffs**

#### **Existing Pre-development Catchments and Flood Hazard**

Section 3.2 of the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18) identified in Section 3.2 nine delineated catchments (A through I) for the pre-developed DNNP site, as shown in Figure 2.5.3.2-1. Information related to catchments A to I are provided in Table 3.2-1 of (Reference 2.5-18), such as area size, land use, soil/surface conditions and runoff. The runoff from Catchment A drains directly into Lake Ontario close to the DNGS forebay. The runoffs from Catchments B, C, D, and E in the north and east flow via the Canadian National Railway right-of-way ditch or through a wetland discharging into the Darlington Creek watershed. The runoffs from Catchments F, G, and H, which are former lay down areas in the DNNP site, flow through culverts southeast of the DWMF building and drain directly into Lake Ontario. The last runoff from Catchment I, a former lay down area, drains through various outlets into Lake Ontario. Potential existing on-site flood hazards include:

- Runoff from Catchments C and D overflowing the Canadian National Railway right-of-way ditch
- Capacity of designed stormwater infrastructure to convey potentially increased peak flows due to proposed DNNP site development.

Subsection 5.4.3 of the 2022 Flood Hazard Assessment (Reference 2.5-18) describes the flood hazard associated with a design basis flood involving PMP falling directly on the DNNP site, assuming 100% impervious land cover. The flood hazard due to direct precipitation is related to the ability of the site development to convey stormwater runoff through the site.

A nodal model (PCSWMM), per Subsection 5.4.3. of the 2022 Flood Hazard Assessment (Reference 2.5-18), of the nine catchments conveyance and retention as well as drainage structures was used to evaluate on-site flood hazards and to size conveyance and retention elements of stormwater for pre-development conditions.

The pre-development results indicate:

- None of Catchments A, B, E, G, and I pose a PMP flood risk on the DNNP site (refer to Table 5.4-11 of Reference 2.5-18)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- Catchments C and D showed significant overflow into the Canadian National Railway right-of-way ditches with no flooding (refer to Table 5.4-12 of Reference 2.5-18)
- While the stormwater infrastructure in Catchment F performs adequately under, for instance, the 1:100-year storm, significant PMP/PMF overflow occurred between its sub-catchments or into neighboring Catchment H, suggesting development is necessary to alter Catchment F and its drainage system (refer to Table 5.4-13 of Reference 2.5-18)
- The stormwater conveyance and retention capacity of Catchment H represents a significant potential overflow under the PMP, between its sub-catchments within the existing infrastructure (refer to Table 5.4-14 of Reference 2.5-18)

These results were carried forward to explore and compare with the post-development results.

*Post-development of BWRX-300 Unit 1 Catchments and Flood Hazard*

A large portion of the pre-development areas of Catchments F and H would be replaced by Catchment N, within which the BWRX-300 Unit 1 footprint would almost entirely be contained, as shown in Figure 2.5.3.2.2. The runoff from Catchment N flows through a series of culverts, roadside ditches, and a pond to a southern outlet into Lake Ontario. The proposed site layout of the BWRX-300 Unit-1 facility will therefore have significant impact on-site catchments and runoff flow directions. Though the upstream reaches of these catchments will still mostly be intact, most of the pre-development of Catchment F and roughly half of Catchment H will be covered by the footprint of the BWRX-300 facility Unit-1 (refer to Figure 3.2-1 in Reference 2.5-18). Conveyance and retention structure of such catchments would consequently require re-configuration.

The same nodal model PCSWMM) was used for post-development conditions including Catchment N. Culvert locations, diameters, conveyance (in  $\text{m}^3/\text{s}$ ) and ditch depths were considered in the assessment. The post-development results for BWRX-300 Unit 1 indicate:

- Catchments A, B, C, D, E, G, and I do not represent a flood hazard for the DNNP site (refer to Subsection 5.4.3.4.1 in Reference 2.5-18)
- Under the PMP, there is significant flooding through the sub-catchments of Catchment F, and to Catchment G (refer to Table 5.4-16 in Reference 2.5-18)
- Current configuration of conveyance and retention structures in Catchment H will experience under the PMP significant flooding into its sub-catchments that may overtop into Catchment G (refer Table 5.4-17 in Reference 2.5-18)
- Catchment N system, comprising ditches, culverts, flood routes and storages, is sized to convey and retain adequately the PMP and is split into 12 sub-catchments described as follows:
  - Sub-catchment N\_1 contains an administrative building and a parking lot and drains south through a culvert into N\_10
  - Sub-catchment N\_2 is a large laydown area, drains through ditch and outlets into N\_12. In the model, all the flow within N\_2 passes through a culvert adjacent to the Power Block, which is a conservative assumption to ensure N\_2 runoff does not flood the Power Block area
  - Sub-catchments N\_3 through N\_7 contain the Power Block area, and each drain through a dedicated culvert into various downstream sub-catchments, with the culvert sizes chosen to ensure zero flooding of the Power Block area
  - Sub-catchment N\_8 is a parking or laydown area draining through a culvert into N\_12

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- Sub-catchment N\_9 is a parking or laydown area draining through a culvert in N\_10
- Sub-catchment N\_10 is a low area adjacent to the Power Block containing a storm water management pond that drains to the south through a culvert into N\_12
- Sub-catchment N\_11 is a low area immediately south of the Power Block accepting flow from N\_5 and N\_7 and conveying through a culvert to N\_12
- Sub-catchment N\_12 is a perimeter ditch, accepting flows from the remainder of Catchment N and conveying toward the Catchment H outlets to Lake Ontario

**Post-development BWRX-300 Unit 1 Modeled Available Freeboard**

The post-development peak flow and flooding results for Catchment N, shown in Table 5.4-18 of (Reference 2.5-18), indicate with “realistic” assumptions (i.e., the largest culvert in the system is 1 m in diameter) for sizing of conveyance and retention structures, the maximum flood level within the Catchment N system is 87.93 m CGD. This provides 0.07 m of freeboard below the 88 m CGD construction grade which is a flood hazard, but by increasing the conveyance and retention capacity of the system, this freeboard can be brought to a comfortable level.

**Comparison of Pre- and Post-development of BWRX-300 Unit 1 Results**

Comparison of pre-developed and post-developed modelling results of BWRX-300 Unit 1 indicate (refer to Subsection 5.4.3.5 and Table 5.4-22 of Reference 2.5-18):

- There are no changes in Catchments A, B, C, D, E, G, and I.
- Maximum flood depth elevation changes between -0.02 m to +0.06 m in Catchment F since it is reconfigured from pre-development conditions.
- Maximum flood depth elevation changes between -0.23 m to +0.17 m occurred in Catchment H since it is also changed in post-development conditions, and it must convey and retain runoff from Catchment F and some runoff that may overtop into Catchment G.

**Impact of Modeling of Four BWRX-300 Units**

Additional modeling analysis showed with proper sizing and arrangement of additional conveyance and retention infrastructure in future site plans, the construction of additional three BWRX-300 units will not impact the functionality of the stormwater infrastructure protecting the first BWRX-300.

**Proposed Flood Mitigation, Proofing, and Practice for DNNP**

In Section 6 of the 2022 Flood Hazard Assessment (Reference 2.5-18), flood mitigation or flood proofing practices applicable to the DNNP as well as mitigation measures are proposed. Options for flood mitigations applicable to the DNNP site include:

- Constructing barriers to stop floodwater from entering the structure/site areas
- Constructing retention and detention ponds to slow and/or stop floodwaters entering the site area
- Wet Flood Proofing whereby floodwaters are allowed to enter the structure/site area, but ensuring that there is no or minimal damage to the building's structure/site and to its contents
- Emergency management/flood forecasting.

Summary of Flood Hazard for the DNNP Site

Table 6.2-2 in the 2022 Flood Hazard Assessment (Reference 2.5-18) summarizes the primary source of flood hazards for the DNNP site due to runoff. In essence, the flood hazards would be to backwatering and flooding of various sub-catchments causing overtopping of the receiving catchments or overloading the existing stormwater management infrastructure. Proposed mitigation includes measures such as:

- Increase the size of specific culverts draining into specific sub-catchments
- Increase the storage capacity of one or more stormwater management ponds
- Route runoff from specific catchments into other specific catchments
- Ensure progressing designs have sufficient conveyance and detention capacity and the stormwater infrastructure is adequate.

Per Subsection 5.4.1 of the 2022 Flood Hazard Assessment NK054-REP-01730-001 (Reference 2.5-18), the PMF, mentioned in Subsection 2.5.3.1, includes a design basis flood (involving a PMP and zero infiltration) concurrent with disabled sewer and drainage systems due to, for example, debris. Therefore, the flooding due to runoff can be screened out based on screening criterion [QL2]. The PMF assessment is the bounding assessment that includes the impacts of potential runoffs.

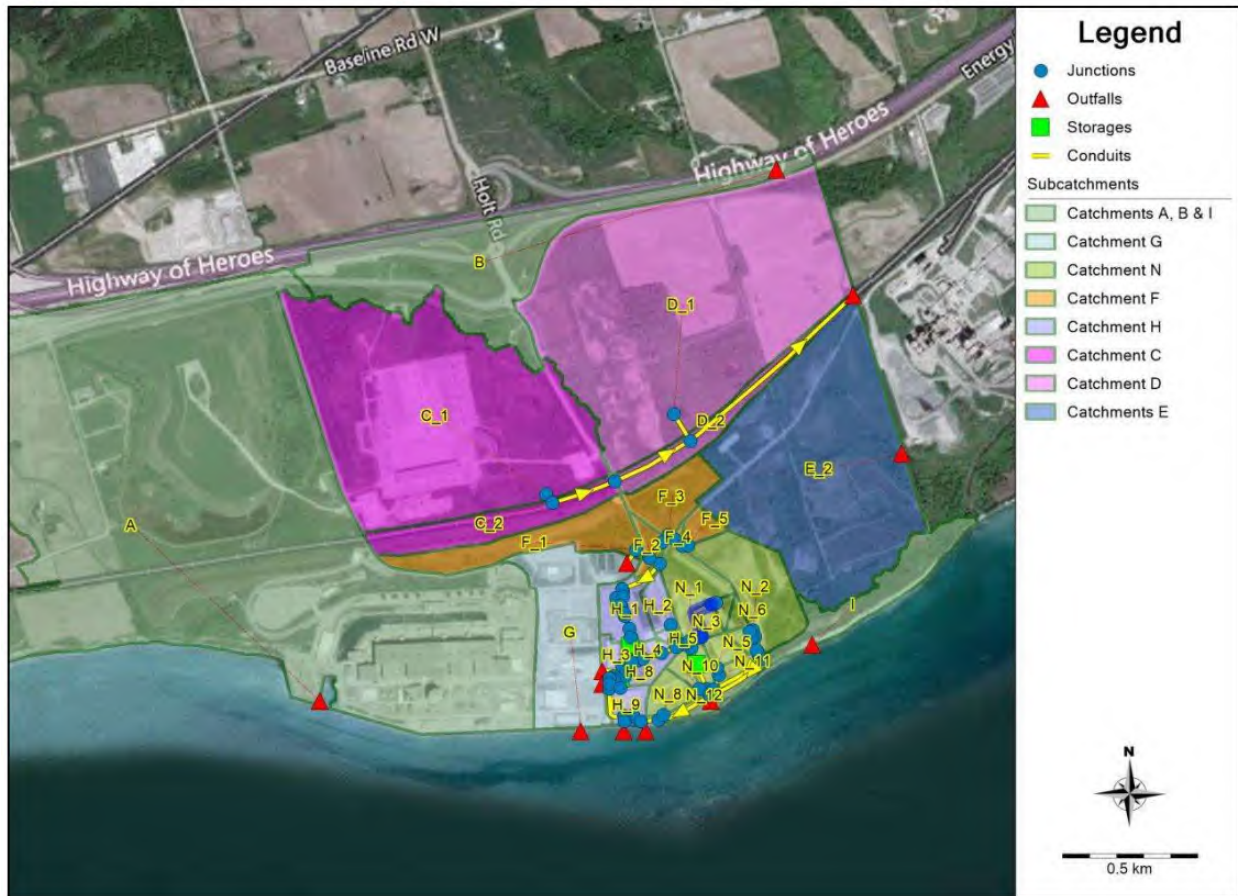
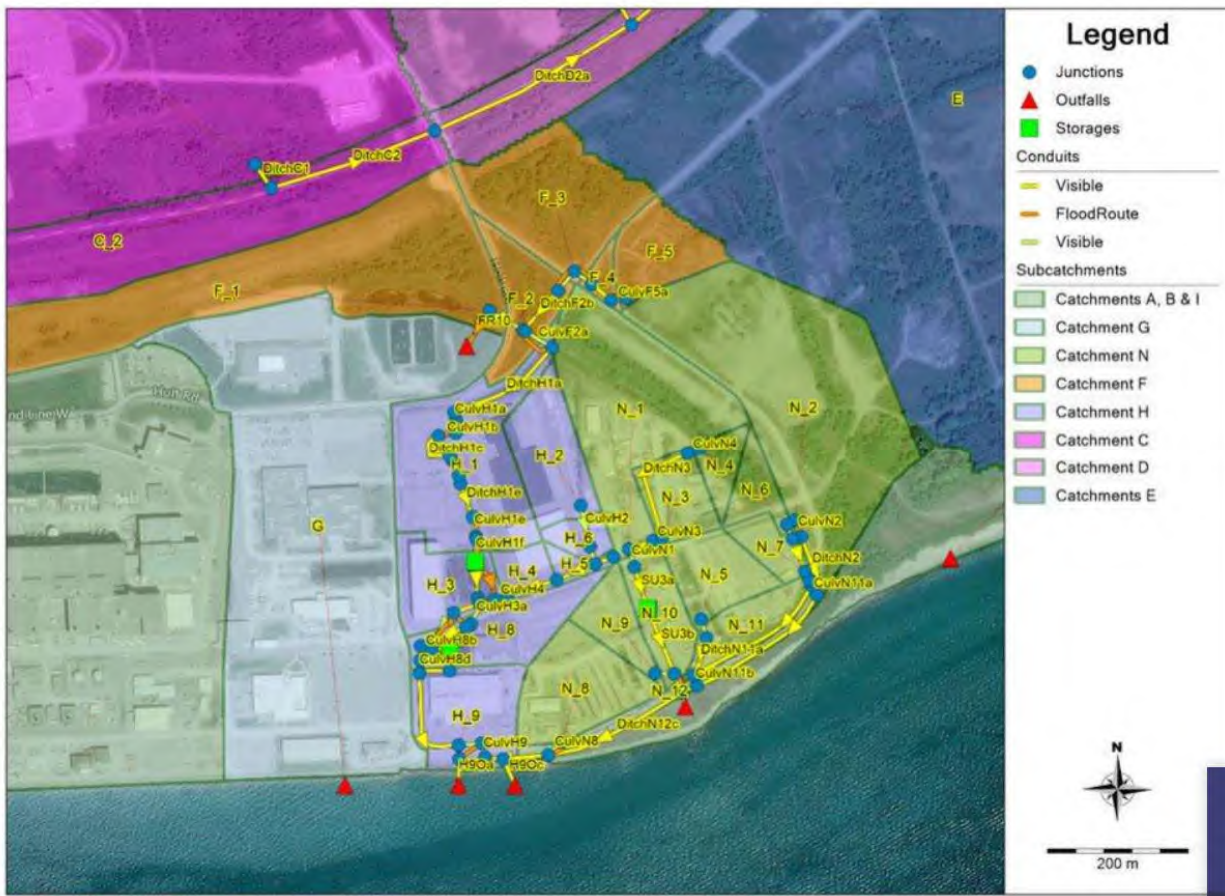


Figure 2.5.3.2-1 Pre-development Darlington Nuclear Site Drainage (Reference 2.5-18)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 2.5.3.2-2 Post-development Darlington Nuclear Site Drainage (Reference 2.5-18)**



### 2.5.3.3 Flooding Due to Rivers

Section 3.1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) names two riverine systems within the local regional drainage basin: Tooley Creek and Darlington Creek.

The distance, infrastructure, and topography between the Tooley Creek watercourse and the proposed DNNP site precludes Tooley Creek as the source of a flood hazard.

Regarding Darlington Creek, the Central Lake Ontario Conservation Authority indicated there is not any history of severe flooding along Darlington Creek within the recorded history of the area. Figure 3.1-6 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) illustrates the current regulatory and 100-year recurrence floods inundation limits and shows that the inundation limits associated with these events do not represent a flood hazard to the DNNP site.

Subsection 5.4.2 in the 2022 Flood Hazard Assessment, NK054-REP-02730-00001 (Reference 2.5-18), describes comprehensive hydrologic and hydraulic models that are used to estimate drainage for the Darlington Creek watershed and its associated 14 sub-watersheds under 100-year recurrence PMP conditions, as replicated in Table 2.5-2. The models considered parameters such as length and slopes of the feeding reaches, time of concentration, storage coefficient, and future 100-year timeframe land use and development. The modelled Darlington Creek flood water elevations under PMP conditions in the future is estimated at 88.5 CGD at a stream gauge cross-section located just south of Highway 401. This is above the DNNP site construction grade of 88 CGD. However, to overtop into the DNNP site, flood waters would have to surpass the lowest elevation along the boundary separating the DNNP site from Darlington Creek, which is 95 CGD. Therefore, no external flood hazard to the DNNP site has been identified from Darlington Creek.

Thus, flooding due to the Tooley Creek and Darlington Creek is screened out for the DNNP site.

**Table 2.5-2 Key Modelling and Assessment Parameters for Darlington Creek and On-site External Flood Hazards (Reference 2.5-18)**

Parameter	Darlington Creek	On-site
Design Storm(s)	2.5-Hour 1:100-Year Storm (4 mm)	12-Hour PMP (420 mm)
	6-Hour PMP (405 mm)	
	12-Hour PMP (420 mm)	
Land Cover	Existing and Future Conditions	Zero infiltration
Threshold Water Level Constituting Flood Hazard	Above 95 m CGD	Above 88 m CGD

### 2.5.3.4 Flooding Due to Waves

The potential for flooding due to waves is discussed in Section 5.3 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2):

1. Subsection 5.3.1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the data and models used to assess the flooding hazard by waves, including the Lake Ontario wind and wave hindcast developed by the Wave Information Studies of the Office, Chief of Engineers, U.S. Army Corps of Engineers. The Simulating Waves Nearshore model was used to propagate extreme wave conditions from a selected 'offshore' wave information studies node to the shoreline, using the SPLASH numerical model for calculations of wave uprush and wave overtopping on shoreline beaches and structures.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2. Subsection 5.3.2 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave hindcast extreme analysis and determines that it is appropriate to use the wave information studies #192 100-year  $H_s$  of 4.7 m with peak wave period  $T_p$  of 9.7 s as input from the SW ( $225^\circ$  N) to wave propagation/overtopping models.
3. Subsection 5.3.3 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave propagation modeling for two water level scenarios and two site layout scenarios.
4. Subsection 5.3.4 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave uprush and overtopping estimates.

Based on these scenarios, the wave uprush estimates range from 3.5 m to 11.3 m, and wave overtopping estimates range from 0.015 to 0.591 m<sup>3</sup>/s/m.

In the 2022 Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.5-18), the calculated wave heights extreme values were based on the latest hindcast data from two stations closest to the DNNP for the period from January 1979 to January 2020. Using a specific fitted method, wave heights were calculated for selected return periods of 10, 50 and 100 years. Based on the results, it was recommended to use an updated design wave of 6.1 m from the SW ( $225^\circ$  N) with peak wave period  $T_p$  of 9.2 s to account for a more conservative estimate of the wave flooding potential at the DNNP site.

#### **2.5.3.5 Flooding Due to Storm Surge and Seiche**

Storm surges may cause seiches, because as a storm moves past the lake, the wind and pressure are no longer pushing the water, therefore the piled-up water moves toward the other end of the lake. The water sloshes from one end of the lake to the other few times until the water level is returned to normal. This sloshing back and forth is called a seiche. Seiches can be created due to other meteorological effects, seismic activities, or also tsunamis.

Section 5.2 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the numerical hydrodynamic model of Lake Ontario which was developed to assess the potential for generation of storm surge and seiche response to extreme severe weather systems tracking through the region. The model was implemented on a bathymetric grid of Lake Ontario with a 2.7 km resolution.

The most severe types of weather systems in the region of Lake Ontario are:

1. Post Tropical Storms: A good example of a post tropical storm with very severe wind conditions for Lake Ontario was Hurricane Hazel (1954). A storm like Hazel would typically approach Lake Ontario from between the southeast and south. A Hazel-like post tropical storm with extremely severe characteristics could have sustained winds up to 100 km/h and a pressure drop as low as 95 kPa.
2. Alberta Clippers: They are compact fast moving winter storms with sustained winds up to about 80 km/h and a pressure drop of about 97 kPa. They would typically track from northwest to west-southwest.
3. Colorado Lows: They are less compact than the Alberta Clippers but have otherwise similar characteristics and would track from the southwest or south-southwest.
4. Gulf Lows: A good example of a very severe Gulf low is the Great Blizzard of 1978. The pressure dropped down to the extremely low value of 95.8 kPa. Characteristic severe sustained winds were up to about 100 km/h.

The parameters used to represent the four idealized storms listed above are shown in Table 5.2-1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2). The highest predicted water level at Darlington Nuclear site resulting from surge or seiche is about 0.75 m. This level can be produced either directly as a surge by a storm of Hazel-type tracking from the south over the western end of the lake, or indirectly after an Alberta Clipper from the west builds up a large surge at the eastern end of the lake resulting in a seiche of large amplitude. The 2022 Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.5-18) also recommended 0.75 m as the highest water level produced by storm surge or seiche, in concurrence with the value predicted in the 2009 NK054-REP-01210-00012 (Reference 2.5-2).

Table 4.2 of the 2019 Darlington Hazard Screening Analysis NK054-REP-03611-1004 (Reference 2.5-4) shows the margin between the lake level and the top of the breakwater works at Darlington Nuclear site. As the margins are larger than the 0.75 m highest water level resulting from surge or seiche, the potential flood impacts are screened out.

#### **2.5.3.6 Flooding Due to Tsunami**

As described in Section 5.7 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2), tsunamis are long period gravity waves generated by seismic disturbances of the sea bottom or shore, or landslides resulting in a sudden displacement of the water surface with the resulting wave energy spreading outwards across the ocean or lake at high speed. An additional consideration is the potential for a tsunami to occur as a series of waves (rather than a single wave) with associated increased impact from cumulative damage or flooding effects.

Due to the geological stability of the Great Lakes region where the largest measured seismic activity results in only small earthquakes typically of magnitude 3 or 4, the 2009 flood hazard assessment NK054-REP-01210-00012 (Reference 2.5-2) concludes a tsunami in Lake Ontario is an improbable event for DNNP. This conclusion is confirmed in the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18).

#### **2.5.3.7 Flooding Due to Ponds, Dams or Dikes**

As noted in Subsection 4.4.7 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4), there is no large permanent human-made water storage pond or dam near the Darlington Nuclear site that can threaten the site. Therefore, this potential flood mechanism is screened out. Per the 2020 NK054-CORR-00531-10533 (Reference 2.5-3), this conclusion is applicable to the DNNP site since it is encompassed by the Darlington Nuclear site. Subsection 5.5.1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) also concluded no hazard assessment for the failure of human-made structures such as dams or dikes is required for the DNNP site.

Any temporary ponds and body of water that could potentially be created during a severe storm (for example on the rail track, by the embankments, overflowing culverts) are addressed in the 2009 hydrological assessment NK054-REP-01210-00012 (Reference 2.5-2) and the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18) (refer to Subsection 2.5.3.2).

#### **2.5.3.8 Flooding Due to Ice Jamming**

As described in Subsection 4.4.8 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4), this event scenario is concerned with late winter conditions when large ice blocks, accumulated over winter, melt rapidly as the weather temperature rises above the freezing point.

The 2014 DNGS hydrological assessment NK38-REP-03611-10094 (Reference 2.5-12) examined the worst-case scenarios and concluded that a summer PMP, with storm drains blocked, would bound winter PMP with snow covering the ground and ice blocking the drains. The event consequences of ice jamming at the lakeshore, and rapid melting of the accumulated

ice blocks may result in localized high water levels and flooding, but the consequences are not worse than the PMF assessed in the DNGS hydrological assessment.

Therefore, the hazard is screened out based on screening criterion [QL2], as both types of consequences (accumulation on the roof tops, and accumulation at the lakeshore) have consequences less severe than the events assessed in the 2014 DNGS hydrological assessment (Reference 2.5-12). This conclusion can be applicable to the DNNP site due to proximity to the DNGS site, per the 2020 NK054-CORR-00531-10533 (Reference 2.5-3).

The 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18) states that in the DNNP site area, Lake Ontario freezing starts from the Bay of Quinte, east of the DNNP site. The ice then propagates eastward to the St. Lawrence River. As shown in Figure 5.6-2 of the 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18), the ice coverage over Lake Ontario is about 17% by mid-February with an average of 10% coverage for the winter period. Ice breaking accelerates in early March. Thus, the DNNP site region of Lake Ontario is ice-free year-round, in an average year. This is mirrored in the fact that, on a weekly basis, between December 4 and May 14, the median ice concentrations near the DNNP site are 0%. Furthermore, Lake Ontario is the smallest Great Lake in terms of surface, but it is the second deepest and as such, has a large volume compared to its surface area, resulting in an exceptionally high heat storage capacity. Temperature changes occur at a much lower rate in Lake Ontario compared to the other Great Lakes.

Therefore, the 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18) confirms that the flood hazard due to ice jamming is screened out based on the basis of screening criterion [QL3].

#### **2.5.4 Potential Effects of Climate Change**

The potential impacts of climate change are discussed and summarized in Subsections 7.2 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18), where Subsections 7.2.1 and 2.7.2 address the effect of climate change on temperature and precipitation.

The total annual precipitations are forecast to slightly increase (+3% to 10%) in 2071-2100 compared to present-day conditions. However, precipitations are expected to remain stagnant during summer, hence resulting in higher percentage increases for other seasons (from +2% to 21%) depending on the emission scenario chosen. Considering that temperature is also forecast to significantly increase during winter, more liquid precipitations are to be expected as well.

Maximum daily precipitations are expected to vary from -4% to +25% depending on season and emission scenario. The seasonal trend follows a similar pattern as total precipitations with stagnant conditions during summer (-4% to 0% compared to present-day conditions) in contrast to spring and winter for example (from +10% to 25%).

Although maximum daily precipitations should not increase by much during fall and especially summer, these seasons remain the period when this extreme weather event will occur. While the projected increase in daily 1:100-year return period precipitation is up to 10.7% by 2100 in the high greenhouse gas emissions scenarios, the PMP evaluated is not anticipated to be exceeded due to climate change, and no additional flood hazard is identified on account of climate change.

Subsection 5.1.2.5 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) describes a 2014 plan that was adopted in 2017 to allow for control of extreme low or high water level conditions. Under the modelled conditions in the 2014 plan, the weekly mean water levels would never have exceeded 75.8 m. However, since the adoption of the new plan in 2017, water level exceeded the previous maximum on two occasions, in 2017 and 2019. Climate change was identified as the probable cause of these maximum water levels.

Subsection 7.2.5 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) discusses the impact of climate change on Lake Ontario water levels. Lake Ontario water levels are primarily controlled by variations in precipitation, runoff, and evaporation over the watershed. Climate change influences these parameters that control lake water level fluctuations. Climate change would contribute to increasing low and high extremes in Lake Ontario water levels. Anticipated increases in precipitation would contribute to high Lake Ontario water levels. The report recommends higher lake levels experienced recently in 2017 and 2019 should be considered as appropriate design lake levels for shoreline assessment and design bases.

According to the 2019 IAEA Site Evaluation for Nuclear Installations Safety Requirements for Flood Hazard (Reference 2.5-14), the reference water level upon which the computed surge or seiche is superimposed should be selected to have a sufficiently low probability of being exceeded. Usually the 100-year recurrence monthly average high water is adopted or, if the water level is controlled, the maximum controlled water level is used. However, the International Joint Commission Lake Ontario 2021 plan (Reference 2.5-13) allows deviations, so that no maximum level is set, and a stochastic approach is still necessary. In this case the controlled water level with a probability of exceedance of 1% is 75.6 m; however, the highest level during a century is about 76.6 m. In addition, measured water levels at Cobourg have exceeded 75.6 m for duration of about three months in 1973.

Therefore, 75.6 m may be a low estimate, and 76.6 m should be used, which is close to the maximum found in the historic data and greater than the 100-year recurrence level. This level assumes the International Joint Commission Lake Ontario continues with the current water level control plan.

The 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts, NK054-PLAN-07007-00001 (Reference 2.5-20) is developed to address the potential impact of climate change on external hydrological and meteorological hazards. The strategy summarizes life cycle considerations including long-term monitoring (Subsection 2.11.9) and describes the plan to ensure the BWRX-300 facility is resilient to climate change as a potential external hazard.

The 2023 NK054-REP-07007-1049426 DNNP Hazard Bounding Analysis (Reference 2.5-22) presents a bounding analysis of climate change impacts and establishes probable extreme values for climate hazards where feasible. The 2022 NK054-REP-07007-1028871 DNNP Gradual Climate Change and Natural Hazard Identification (Reference 2.5-23) describes the process used in identifying a comprehensive list of natural external events for DNNP, which are screened for climate change impact for evaluation against the DNNP BWRX-300 design basis.

## **2.5.5 Groundwater**

Relevant to the assessment of radioactive material transported through the groundwater system and potentially dispersed in the environment, the following subsections discuss the characterization of the hydrogeological subsurface properties as well as relevant monitoring programs.

The in-situ soil properties are derived based on existing subsurface investigations completed at the DNNP site and in the vicinity of the BWRX-300 SMR location, as described in Subsection 2.7.3.2.4.

### **2.5.5.1 Groundwater Conditions**

The groundwater conditions are described in detail in Subsection 2.7.3.2.4. Groundwater flow maps are available in Section 2.7, Figures 2.7.3.2-3 to 2.7.3.2-9. In general, groundwater on the site flows from north to south, and discharges toward Lake Ontario, as confirmed in the 2022 DNNP Phase 1 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.5-21).

The predominant groundwater flow patterns reported in the 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.5-21) remain unchanged from the historical interpretations of groundwater flow conditions documented in the 2009 NK054-REP-01210-00011 (Reference 2.5-15) and the 2009 NK054-REP-07730-00005 (Reference 2.5-16).

Relevant information is provided in Subsection 2.8.2.2 on the impact of hydrogeological conditions on the dispersion of radioactive material.

#### **2.5.5.2 Groundwater Level**

Based on the groundwater conditions at the DNNP site presented in Subsection 2.7.3.2.4 and Table 2.7-11, groundwater is anticipated to be present approximately between elevation 80 m to 86 m corresponding to depths between about 2 m and 8 m below the plant grade at elevation 88 m. (refer to Subsection 2.7.5.2.6)

#### **2.5.5.3 Groundwater Monitoring**

The OPG Environmental Monitoring Program (EMP) N-REP-03443-10027 (Reference 2.5-17) examines the chemical, radiological, and physical characteristics of the groundwater beneath the Darlington Nuclear site. The groundwater monitoring wells are located in key areas of the Darlington Nuclear site including the protected areas (near the RBs), controlled areas (farther away from the RBs but within the fence), and the Darlington Nuclear site perimeter. Wells on DNNP site are considered site perimeter wells (refer to the NK38-REP-10140-10032 (Reference 2.5-8)).

#### **2.5.6 Surface Water**

As related to the assessment of radioactive material transported through the surface water system and potentially dispersed in the environment, this subsection discusses the characterization of the surface water properties in Subsection 2.5.6.1, as well as the relevant monitoring programs in Subsection 2.5.6.2.

##### **2.5.6.1 Surface Water Properties**

The pertinent properties of the surface water (i.e., Lake Ontario) are described below:

###### **1. Lake-Wide Circulation**

The Darlington Nuclear site is situated on the northern shore of Lake Ontario where the lake-wide circulation is generally eastward from the Niagara River to the discharge to the St. Lawrence River, per the 2021 D-REP-07701-00001 (Reference 2.5-9). Water movement near the site is predominantly along the shore, occurring for 73% of the time (35% to the west and 38% to the east), as described in the 2012 NK054-REP-01210-00016 (Reference 2.5-5). Onshore and offshore movement occurs about 15% of the time, as reported in the 2012 NK054-REP-01210-00016 (Reference 2.5-5). Table 2.7 in the 2021 D-REP-07701-00001 (Reference 2.5-9) shows the frequency of lake current flowing toward each direction and the maximum speed that occurred in each direction, per the 2021D-REP-07701-00001 (Reference 2.5-9). Table 2.5-3 shows the averaged lake current direction and speeds.

**Table 2.5-3 Summary of Lake Ontario Depth Averaged current speed and direction  
(Reference 2.5-9)**

Month	Direction	Depth Averaged Speed All Directions	Depth Averaged Speed Easterly	Depth Averaged Speed Westerly
	Degree from North	cm/s	cm/s	cm/s
January	142	17.5	20.6	12.4
February	145	16.2	18.9	13.1
March	159	13.5	15.5	12.7
April	165	11.8	12.7	12.3
May	181	9.4	12.0	7.8
June	177	9.5	10.5	9.7
July	183	13.3	16.0	11.4
August	193	10.9	12.2	11.1
September	196	9.9	10.3	10.9
October	170	11.8	13.0	11.9
November	159	11.5	13.2	9.8
December	169	12.9	14.4	12.5
Annual Average		12.4	14.1	11.3

**2. Lake Water Temperature**

Lake Ontario is classified as a dimictic lake because it undergoes a complete cycle of isothermal and vertically stratified conditions every year. The thermal structure depends on the season because of large annual variation in surface heat fluxes. Lake-wide surface temperatures typically range from freezing in winter to about 20 °C in summer, per the 2021 D-REP-07701-00001 (Reference 2.5-9). Statistical summary of ambient water temperatures near Darlington Nuclear site (from 1984 to 1996 and 2011 and 2012) is provided in Table 2-9 of the 2021 D-REP-07701-00001 (Reference 2.5-9).

**3. Ice Conditions**

Ice formation in winter is typically limited to the nearshore areas at the eastern end of the lake within the Kingston Basin, per the 2021 D-REP-07701-00001 (Reference 2.5-9) and the 2022 NK054-REP-02730-00001 (Reference 2.5-18).

**2.5.6.2 Surface Water Monitoring**

As described in Subsection 3.2.2 of the 2019 NK38-OM-61100 (Reference 2.5-6), the Lake Current Monitoring system is a real-time current profile measurement system to be used in the event of a radiological liquid emission. Further details of the radiological baseline conditions of lake water at the Darlington Nuclear site are provided in Subsection 2.9.1.1.

The OPG EMP N-REP-03443-10027 (Reference 2.5-17) identifies the contaminants and physical stressors to be monitored in the environment surrounding the site. Locations considered to be outside the influence of site operations are also monitored to allow for a baseline comparison with

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

background values. This includes monitoring and sampling of lake water, municipal drinking water, and other means of aquatic sampling. Further details on the EMP are provided in Chapter 20, Subsection 20.11.2.

### **2.5.7 References**

- 2.5-1 NK054-REP-01210-00018 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Additional Considerations," Ontario Power Generation.
- 2.5-2 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.5-3 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation Licence (PRSL)," Ontario Power Generation.
- 2.5-4 NK054-REP-03611-10043 R003, 2019, "Hazard Screening Analysis – Darlington," Ontario Power Generation.
- 2.5-5 NK054-REP-01210-00016 R002, 2012, "Site Evaluation of the OPG New Nuclear at Darlington - Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.
- 2.5-6 NK38-OM-61100 R013, 2019, "Environmental Monitoring – Air and Water," Ontario Power Generation.
- 2.5-7 NK054-REP-01210-00108 R000, 2019, "DNNP – Site Preparation Nuclear Safety Licence Renewal Activity Report," Ontario Power Generation.
- 2.5-8 NK38-REP-10140-10032 R000, "Darlington Nuclear Groundwater Monitoring Program Results," Ontario Power Generation.
- 2.5-9 D-REP-07701-00001 R001, 2021, "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.5-10 IAEA Safety Standards No. SSG-18, 2011, "Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations," International Atomic Energy Agency.
- 2.5-11 "Lakes, and Rivers Improvement Active Technical Guidelines Administrative Guide," 2017, Ministry of Natural Resources and Forestry.
- 2.5-12 NK38-REP-03611-10094 R000, 2014, "Darlington Nuclear Generating Station Hydrological Assessment," Ontario Power Generation.
- 2.5-13 International Joint Commission Lake Ontario, "St. Lawrence River Water Levels, June 2021," <https://ijc.org/en/loslrb/watershed/water-levels>.
- 2.5-14 IAEA Safety Standards Series No. SSR-1, 2019, "Site Evaluation for Nuclear Installations Safety Requirements," International Atomic Energy Agency.
- 2.5-15 NK054-REP-01210-00011 R001, 2009, "Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.5-16 NK054-REP-07730-00005 Rev. R000, 2009, "Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.5-17 N-REP-03443-10027 R000, 2021, "Results of Environmental Monitoring Programs," Ontario Power Generation.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- 2.5-18 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.
- 2.5-19 NK054-REP-02730-00002, 2022, "PMP Validation," Ontario Power Generation.
- 2.5-20 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation.
- 2.2-22 NK054-REP-01210-00175 R000, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation
- 2.5-22 NK054-REP-07007-1049426 R001, 2023 "Darlington New Nuclear Project – Hazard Bounding Analysis," Ontario Power Generation
- 2.5-23 NK054-REP-07007-1028871 R000, 2022 "Darlington New Nuclear Project – Gradual Climate Change and Natural Hazard Identification," Ontario Power Generation



## **2.6 Meteorology**

### **2.6.1 Introduction**

Section 2.6 describes the meteorological aspects relevant to the DNNP site based on the consideration of the local climatic effects. Details are included in Section 2.6 on the characterization of extreme values of meteorological events in relation to potential hazards to the BWRX-300 facility, as well as in relation to the transportation of radioactive materials and the dispersion of radionuclides with the potential to impact the DNNP site. The meteorological characteristics and conditions included in the following list are assessed in relation to the design and the evolution of extreme parameters over the lifetime of DNNP BWRX-300:

- Temperature (Subsection 2.6.2)
- Humidity (Subsection 2.6.3)
- Rainfall (Subsection 2.6.4)
- Wind Speed (Subsection 2.6.5)
- Tornadoes and Hurricanes (Subsection 2.6.6)
- Waterspouts (Subsection 2.6.7)
- Dust Storms and Sandstorms (Subsection 2.6.8)
- Snow Load and Ice Load, Freezing Rain, and Ice Storm (Subsection 2.6.9)
- Lightning (Subsection 2.6.10)
- Windborne Debris (Subsection 2.6.11)
- Climate Change (Subsection 2.6.12)

Key metrological characteristics and parameters relevant to the DNNP site and the surrounding area are listed in Table 2.6-1. The list includes characteristics such as temperature, humidity, precipitation, high wind, tornadoes, snowfalls, lightning, and climate change impact.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area**

Characteristic	Value/Description				
2.6.2 Temperature	Highest ever recorded	Toronto Bowmanville		36 °C 40.6 °C	
	Extreme minimum	-40 °C, with annual degree-days below 18 °C of 4130 degree-days			
	Maximum	Dry bulb 37 °C		Wet bulb 23 °C	
	Design Basis Duration at low Temperature		Temperature	Duration	
			-40 °C	1 h	
			-35 °C	5 h	
			-30 °C	10 h	
			-25 °C	20 h	
			-20 °C	70 h	
	-15 °C	150 h			
Safety Class 1 SSC Design Conditions	Highest 40 °C		Lowest -40 °C		
Impact of extreme temperatures	Mist and white frost during winter		Heatwaves during summer		
Impact of Climate Change by 2100	Increase between 2 °C and 5 °C (References 2.6-3 and 2.6-4) Recent analysis: increase by up to 7.2 °C (Reference 2.6-17)				
2.6.3 Humidity	Lowest	During winter and air is quite dry due to Arctic air from the north			
	Highest	During summer and fall due to the air from the Gulf of Mexico.			
	Mean value	65 to 80% throughout the year			
	Design Conditions	No indication of extreme conditions that require design mitigation			
2.6.4 Rainfall / Precipitation	Mean annual	Oshawa 877.9 mm		Toronto 800 mm	
	Maximum daily	Oshawa 88.6 mm		Toronto 79.3 mm	
	Average (DNGS PO-SAR)	145 days/yr, with of 800 mm average, with 20% due to snowfall			
	Greatest per day	In Oshawa, 144.8 mm			
	PMP (vicinity of DNNP)	420 mm in 12-hours, with 51% in the 6 <sup>th</sup> hour, for a watershed area of < 1295 km <sup>2</sup>			
	Severe flooding	PMP conditions, combined with a 1 in 100-year recurrence lake level high, and storm surge			
	Extreme Daily	Unlikely to exceed the PMP value in a 100-year recurrence for DNNP			
	For roof design	16 mm in 5 min – 50-year, 5-minute storm 25 mm in 15 min - 50-year return, 15-Minute storm 47 mm in 60 min – 50-year return 1-hour storm 210 mm in 24 h – Regional storm (Hurricane Hazel)			
	Climate Change Impact by 2100	Increase in heaviest precipitation intensity and frequency of 12% and 22%, respectively. Plausible increase in extreme precipitation amount over southern Ontario by 14% (7 mm) (Reference 2.6-3).			

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area**

Characteristic	Value/Description	
		Recent analysis indicates total precipitation and maximum 24-hour re anticipated to increase by up to 25% (Reference 2.6-17). The 12-hour PMP of 420 mm remains bounding of this increase as the summer and fall projections (when PMP would occur) are lower, up to +10%, and the PMP value is conservative (Reference 2.6-18). Such predicted changes is to be considered in the design and monitored for long term as discussed in Subsection 2.11.9.
<b>2.6.5 Wind and Wind Speed</b>	Typical	The prevailing winds were from the north-westerly quarter (10.38% of the time) and from the west quarter (9.98% of the time) (Refer to Subsection 2.8.1.3)
	Average and Clam	Approximately 2.4 m/s (~8.6 km/h) and less than 2 m/s (~7.2 km/h), respectively at 10 m level (Refer to Subsection 2.8.1.3)
	Maximum	64 km/h at 10 m level and 80 km/h at 50 m level (for a 100-year return period)
	Wind 3-sec Gust	Extreme gusts – Occur mostly in the West, Southwest, and Northwest directions Speeds exceeding 120 km/h are rare Higher speeds of up to 174.4 km/h occurred in some instances
	Climate Change Impact by 2100	Wind speeds are expected to change due to climate change. Decline in average wind speed over the years in a warmer world
<b>2.6.6 Tornadoes and Hurricanes</b>	Maximum Pressure Drop	6.3 kPa
	Maximum Rotational Speed	257.4 km/h
	Maximum Transitional Speed	64.4 km/h
	Maximum Wind Speed	321.8 km/h (Upper limit - Enhanced Fujita scale 4 (EF-4) tornado)
	Radius of Maximum Rotational Speed	45.7 m
	Rate of Pressure Drop	2.5 kPa/s
	Design Basis – Tornado Missile Spectrum types	Schedule 40 pipe, Automobile 5 m x 2 m x 1.3 m, and Solid Steel Sphere (Refer to Table 2.6-6)
	Hurricanes, Cyclones, Tropical Storms, Tropical Depression	Very low probability of an actual hurricane directly impacting the DNNP site, and it describes the probable maximum tropical cyclone as unlikely to yield gusts of more than 100 km/h - lower than that of the design basis tornado. As such, wind hazard from a hurricane is not considered further.
<b>2.6.7 Waterspouts</b>	A tornado that forms over water that are rarely reported. Covered by the design basis tornado	
<b>2.6.8 Dust and Sandstorms</b>	Not identified as phenomena for southern Ontario, and as such are not identified as potential hazards for DNNP.	
	Average daily snowfall	3 cm to 5 cm from December to March

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area**

Characteristic	Value/Description	
<b>2.6.9 Snow and Ice Load, Freezing Rain, and Ice Storm</b>	Highest Daily snowpack	Mean value of 8.6 cm in January
	Darlington Nuclear site characteristic Value	2.2 kPa
	Combined snow load and winter PMP event	1.80 kPa for 50-year recurrence 1.71 kPa for 100-year recurrence, without Winter PMP
	Freezing Rain	Screened out due to low frequency
	Ice Storm	This issue is resolved as part of Pressure Increase Group (refer to Subsection 2.2.8).
<b>2.6.10 Lightning</b>	Frequency	2 to 3 cloud-to-ground flashes per year per square km, causing induced fires and electromagnetic compatibility. Screened out due to low hazard to the site.
<b>2.6.11 Windborne Debris</b>	Wind-propelled missiles are similar to tornado missiles which is assessed as part of the high wind PSA.	
<b>2.6.12 Climate Change Impact</b>	Impact of climate change is considered in the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-REP-07007-00001 (Reference 2.6-19) which summarizes life cycle considerations including long-term monitoring, described in Subsection 2.11.9	

## 2.6.2 Temperature

Since DNNP is in proximity to DNGS within the Darlington Nuclear Site, similar meteorological conditions are expected. The highest temperatures ever recorded at Bowmanville, and Toronto are 36 °C and 40.6 °C, respectively per Subsection 2.1.1 of the 2019 NK054-REP-01210-00108 (Reference 2.6-2). As shown in Table 2-1 of Part 2 of the 2018 NK38-SR-03500-10001 DNGS Safety Report (Reference 2.6-7), the extreme minimum temperature chosen for DNGS was -40°C, with annual degree-days below 18 °C of 4130 degree- days. Per Subsection B.8.4. Table 3 of the 2010 N-REP-01200-10000 (Reference 2.6-9), the Darlington Nuclear site characteristic value for maximum dry bulb temperature is 37°C, and the maximum wet bulb temperature is 23°C. The design basis durations at low temperature for DNGS site in the 2018 NK38-SR-03500-10001 (Reference 2.6-7), which are applicable to the DNNP site, are listed in Table 2.6-2.

**Table 2.6-2: DNGS Design Basis Durations at Low Temperature Applicable to DNNP**

Temperature	Duration
-40°C	1 h
-35°C	5 h
-30°C	10 h
-25°C	20 h
-20°C	70 h
-15°C	150 h

According to Subsection 4.5.1 of the 2012 NK054-REP-01210-00016 (Reference 2.6-6), Safety Class 1 (SC1) SSCs that are exposed to ambient environment conditions in DNGS are designed for extreme temperatures of -40 °C during the winter and +40 °C during the summer. The design temperature for the DNNP SSCs is -40 °C, while the design temperature of +40 °C is approximately the same as the highest recorded temperature of 40.6 °C as baseline data on extreme conditions. Although the HVAC system efficiency is generally reduced due to extreme high temperature conditions, the system is expected to provide sufficient cooling to maintain design limits for equipment rooms and to support control rooms habitability. This information is also relevant to DNNP SSCs which require the implementation of appropriate mitigating measures, as necessary.

Refer to Chapter 9A, Section 9A.5 for information on the functions, design bases, description, maintenance, performance, and safety evaluation of the BWRX-300 HVAC systems.

Furthermore, global climate models projected in 2009 an increase of the temperatures in southern Ontario of between 2 °C and 5 °C over the next century, due to rising greenhouse gas emissions, as indicated in Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3). This information is in line with the contents in Subsection 4.1.2.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4), which stated temperatures in the vicinity of DNNP site were expected to rise by 2 °C in 2040 and by as much as 5 °C in 2100 during winter and summer months. In the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.6-17), Subsection 7.2.3 indicates temperatures at the DNNP site are anticipated to increase by up to 7.2 °C by 2100. Mitigation of these environmental changes is to be provided at DNNP. Subsection 2.11.9 describes the long-term monitoring of parameters susceptible to be impacted by climate change,

as mentioned in the 2023 DNNP Strategy for Addressing Climate Change Impacts, NK054-PLAN-07007-00001 (Reference 2.5-20).

The extreme temperatures expected in the vicinity of DNNP site have the potential to result in mist and white frosts during winter, and heatwaves during summer, per Subsection 4.5.1 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1). In the event of extremely high temperature conditions, an extended heatwave could lead to a high demand on the transmission lines, which could potentially cause a loss of grid condition.

Temperatures Normals at and near the Darlington Nuclear site are described in Subsection 2.8.1.1, as related to the meteorological impact on the dispersion on radioactive material.

### **2.6.3 Humidity**

The 2009 Site Evaluation of Meteorological Events NK054-REP-01210-00013 (Reference 2.6-4) states the average relative humidity in the vicinity of DNNP is the lowest during winter, as the air is quite dry due to the Arctic air moving down from the north; the highest humidity values occur during summer and fall as the humid air from the Gulf of Mexico moves across southern Ontario.

Currently, humidity values are not recorded on-site by the meteorological tower as indicated in Subsection 2.2.1 of the 2012 NK054-REP-01210-00016 (Reference 2.6-6). However, this information is available from several Environment Canada stations such as Oshawa WPCP and Toronto Island. Based on the available data, the mean relative humidity ranges from 65 to 80% throughout the year, per Section 2.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5). Section 3.11 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5) also states the meteorological values evaluated with respect to humidity show no indications of extreme conditions requiring design mitigation. Based on Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8), no further evaluation is required on the impact of humidity, as the design of the BWX-300 facility is expected to fit within the Plant Parameter Envelope (PPE) values per commitment D-C-3 in the 2021 NK054-REP-01210-00078 DNNP Commitments Report (Reference 2.6-10).

### **2.6.4 Rainfall / Precipitation**

The Bowmanville Mosert climate station is the closest to the Darlington Nuclear site. The Precipitation Normals (from 1981 to 2010) are described in Subsection 2.8.1.2, where the monthly averages and daily extremes (for precipitation (mm), rain (mm), and snow (cm)) are listed in Table 2.8-3.

The concept of PMP is defined in the 2009 NK054-REP-01210-00012 (Reference 2.6-3) as the greatest depth of precipitation possible for a given storm area at a particular location and time of the year (refer also to Subsection 2.5.3.1 for details on PMP and PMF definitions and values). According to Section 4.1 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), the PMP for watershed areas less than 1295 km<sup>2</sup> in the vicinity of DNNP site has been estimated as a 12-hour precipitation equivalent to 420 mm of total rainfall (with 51% in the 6th hour). Hence, based on the maximum daily precipitation predicted in Subsection 3.4.3 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) using data from the monitoring stations in Toronto Island and Oshawa (79.3 mm and 88.6 mm, respectively), it is unlikely for extreme daily precipitations to exceed the 420 mm PMP value in a 100-year period for DNGS. This conclusion, which is also applicable to DNNP given its proximity to DNGS, is confirmed in the 2022 DNNP Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.6-17).

Precipitation, along with other meteorological factors such as wind direction and speed, influence dispersion and, in case of precipitation, especially deposition. Radioactive materials tend to flow toward low-pressure systems and rainfall often occurs around those systems. Having the PMP

value for the DNGS site available for the DNNP PPE ensures that this (maximum probable) value is considered in the DNNP's dispersion (and deposition) models. Models/codes (such as ADDAM and PAVAN) would use the precipitation rate as input to wet deposition. Precipitation Normals at and near the Darlington Nuclear site are described in Subsection 2.8.1.2, as related to the meteorological impact on the dispersion on radioactive material.

According to Table 3-1 in the 2022 PMP Validation reported in NK054-REP-02730-00002 (Reference 2.6-18), the DNNP storm values to be considered as part of roof design are as follows:

- 210 mm in 24 h – Regional storm (Hurricane Hazel)
- 47 mm in 60 min – 50-year return 1-hour rainfall
- 25 mm in 15 min - 50-year return 15-Minute storm
- 16 mm in 5 min – 50-year 5-minute storm

In relation to the changes in precipitation over time, few studies have examined changes in precipitation over Canada. The 2009 site evaluation report on flood hazard assessment, NK054-REP-01210-00012 (Reference 2.6-3) provides references to a number of studies in Subsection 7.2.1. Based on the conclusions in this report, the heaviest precipitation events are becoming more frequent during the spring and summer, and less frequent during the winter. The information provided indicates a reported increase in extreme precipitation intensity and frequency of 12% and 22%, respectively. In addition, Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3) states some models show a plausible increase in the amount of precipitation for the most extreme precipitation events over southern Ontario by 14% (7 mm). In the 2022 NK054-REP-02730-00001 DNNP Flood Hazard Assessment (Reference 2.6-17), Subsection 7.2.3 indicates the total precipitation and the maximum 24-hour for certain seasons to increase by up to 25% by 2100. The PMP event is not coincident with this increase and remains conservative considering anticipated coincident increases. Consequently, no additional flood hazard is considered for rainfall increase due to climate change. However, as discussed in Subsection 2.11.9, long-term monitoring of precipitation is included as part of the 2023 DNNP Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.6-19).

### **2.6.5 Wind and Wind Speed**

Wind data sets at a standard height of 10 m are collected from Darlington Nuclear site meteorological tower as well as from nearby monitoring stations. The Darlington Nuclear site average and calm wind speeds and wind direction data are presented in Subsection 2.8.1.3. The maximum wind speed at 10 m level and 50 m level at Darlington Nuclear site was estimated to be 64 km/h and 80 km/h, respectively, for a 100-year return period, per the 2009 NK054-REP-01210-00013 (Reference 2.6-4).

Wind gust analysis is performed in the 2022 NK0054-REP-02730-00003 (Reference 2.6-14) for the DNNP site. Although wind speed was collected at the DNGS for 12 years at 15-minutes intervals, 3-second wind gust data were not available. In the 2022 NK0054-REP-02730-00003 (Reference 2.6-14) high-quality Government of Canada publicly available 3-second wind gust data were used from four different stations located within 100 km from DNNP: the three airports in Toronto, Peterborough, and Trenton, as well as the Toronto City Centre. Wind roses were used to analyze the gust magnitude and frequency for each station in eight gust directions. Annual Maximum Series data were then extracted and statistically tested and analyzed. Based on the summary of the maximum and mean of gust Annual Maximum Series, extreme gusts were found to occur mostly in the West, Southwest, and Northwest directions.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

To estimate the extreme design gust speeds for various return periods, the Extreme Value Type I model (known as Gumbel distribution model) was fitted to the extracted Annual Maximum Series values, as described in the 2022 NK0054-REP-02730-00003 Wind Gust Analysis (Reference 2.6-14). The extreme design gust speeds were then calculated for various return periods, particularly, for the design of reactor buildings based on ASCE7 IV risk category which corresponds to 3000-year return period. Other commonly used values corresponding 300-, 700-, and 1700-year return periods were also estimated. Finally, Inverse Distance Weighted interpolation technique was applied to transfer the estimated 3-second gust values from the four selected stations to the DNNP site; the results are listed in Table 2.6-3. Also, bounding envelop 3-second gust extreme values were computed for the DNNP site, as listed in Table 2.6-4. The envelop values are found to be on average 6% higher than the values estimated through interpolation for the DNNP site. Hence, for the design to be conservative, the 2022 NK0054-REP-02730-00003 Wind Gust Analysis (Reference 2.6-14) recommends using the envelop values.

Table 3-5 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5) presents the historical data available for wind gusts in the nearby area to the Darlington Nuclear site. Similar to the methodology used in Subsection 3.4.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) and the 1990 N-REP-NGD-IR-61100-0002 (Reference 2.6-11), site-specific 3-second gust wind speeds of more than 120 km/h or more are rare. However, 3-second gust wind speeds have occurred in some instances with a maximum historical wind gust in the area of 154 km/h. This is consistent with Table 4-4 of the 2022 NK0054-REP-02730-00003 (Reference 2.6-14), as presented in Table 2.6-3, noting maximum speeds of up to 174.4 km/h occurred in some instances.

[The hazards associated with high winds were not addressed in the 2019 DNGS hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1). However, there is a commitment in place by OPG to perform a high wind PSA as part of the Licence to Construct application, as indicated in Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8). The high wind PSA will consider the impact from wind pressure-loading effects and wind-propelled missile analysis from various categories of high wind, and their impact on severe core damage and large release analysis.

The review of literature and simulations from Environment Canada indicated in Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3) points to expected changes in wind speed due to increased greenhouse gas emissions. The same subsection states the global average winds are expected to decrease in a warmer world due to the decrease in temperature differential between the equator and poles. In the 2022 NK054-REP-02730-00003, Wind Gust Analysis (Reference 2.6-14), it was reported that Lake Erie shores will experience a decrease in wind speeds of 5% by 2071-2100, while other areas in Ontario like James Bay and Georgian Bay will experience an increase in wind speeds ranging from 1.4% to 10%.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.6-3: Extreme 3-second Gust Speeds for the DNNP (Reference 2.6-14)**

	Return Period (year)	Gust speed (km/h) for each direction							
		NE	E	SE	S	SW	W	NW	N
<b>DNNP Site</b>	3000	116.5	153.4	106.2	131.7	172.3	165.6	145.5	115.8
	1700	115.5	147.7	101.5	125.7	165.3	159.5	140.5	111.3
	1000	106.7	142.0	97.4	120.2	158.8	153.9	136.0	107.3
	700	103.8	138.9	94.6	116.7	154.6	150.3	132.9	104.5
	300	96.5	130.5	87.9	108.0	144.3	141.4	125.6	97.9
	200	93.0	126.0	84.7	103.6	139.6	137.2	122.0	94.6
	100	87.0	120.0	79.1	96.8	131.0	129.7	116.0	89.4
	50	81.0	113.0	73.6	89.7	122.5	122.5	110.0	83.9
	20	72.9	103.0	66.2	80.2	111.5	112.9	102.0	76.8
	10	66.7	96.3	60.6	72.8	102.5	105.2	95.8	71.3

**Table 2.6-4: Extreme 3-second Gust Speeds Envelop Based on Four-station Data Around the DNNP (Reference 2.6-14)**

	Return Period (year)	Gust speed (km/h) for each direction							
		NE	E	SE	S	SW	W	NW	N
<b>DNNP Site</b>	3000	123.3	153.4	121.4	135.2	174.4	170.7	145.5	143.1
	1700	118.1	147.7	116.0	129.0	167.6	164.6	140.5	137.7
	1000	113.0	142.0	111.0	123.0	161.4	159.0	136.0	133.0
	700	110.2	138.9	107.8	119.6	157.2	155.4	132.9	129.3
	300	102.6	130.5	100.0	110.6	147.3	146.6	125.6	121.3
	200	99.0	126.0	96.3	106.0	142.5	142.0	122.0	117.0
	100	92.8	120.0	89.8	99.0	134.4	135.0	116.0	111.0
	50	86.5	113.0	83.4	91.6	126.2	128.0	110.0	104.0
	20	78.2	103.0	74.8	81.7	115.2	118.0	102.0	95.6
	10	71.8	96.3	68.1	74.1	106.8	111.0	95.8	88.8

## 2.6.6 Tornadoes and Hurricanes

### Tornadoes

As discussed in Section 3.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5), tornadoes are characterized as a rare and non-negligible threat, and a study of a design basis tornado was conducted to estimate the probability of occurrence at the DNNP site. The results of this study are presented in Table 3-7 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5).

The DNNP site characteristics associated with the design basis tornado are outlined in Table 3 of the 2010 N-REP-01200-10000 (Reference 2.6-9), summarized in the following, and listed in Table 2.6-5 and Table 2.6-6):

1. Maximum Pressure Drop – The design assumption for the decrease in ambient pressure from normal atmospheric pressure due to the passage of the tornado
2. Maximum Rotational Speed – The design assumption for the component of tornado wind speed due to the rotation within the tornado
3. Maximum Translational Speed – The design assumption for the component of tornado wind speed due to the movement of the tornado over the ground
4. Maximum Wind Speed – The design assumption for the sum of maximum rotational and maximum translational wind speed components
5. Radius of Maximum Rotational Speed – The design assumption for distance from the centre of the tornado at which the maximum rotational wind speed occurs
6. Rate of Pressure Drop – The assumed design rate at which the pressure drops due to the passage of the tornado
7. Tornado Missile Spectra – The design assumptions regarding missiles that could be ejected either horizontally or vertically from a tornado. The spectra identify mass, dimensions, and velocity of credible missiles

The DNNP site characteristics values in the 2010 N-REP-01200-10000 (Reference 2.6-9) are based on the U.S. NRC Regulatory Guide 1.76 Rev 1 (Reference 2.6-13), Region 2 design basis tornado values. The characteristics, and appropriate reasoning are summarized from the 2022 NK054-CORR-01210-1015770 Engineering Direction for DNNP Design Basis Tornado Values (Reference 2.6-12). The DNNP site is conservatively assumed to have the Site Characteristic Maximum Wind Speed Site Characteristic value of 321.8 km/h for maximum wind speed. This is supported by the following reasons:

- The Maximum Wind Speed of 321.8 km/h is the upper limit for an Enhanced Fujita scale 4 (EF-4) tornado.
- The Maximum Wind Speed of EF-4 is a conservative value for the Darlington Nuclear site, as the Maximum Wind Speed value is not a measured value for the site.
- The assessment performed of the occurrence of tornadoes within an area of 100 000 km<sup>2</sup> of the Darlington Nuclear site during the past 50 to 60 years indicated two category Enhanced Fujita scale 4 (EF-4) tornadoes were observed within 180 km of the site during that period.
- A probability of approximately 0.01% per year was predicted corresponding to an EF-4 category of damage for the Darlington Nuclear site.
- The U.S. NRC RG-1.76 Rev1 (Reference 2.6-13) values for the two subregions adjacent to the Eastern Great Lakes and the northeastern boundary of Region 1 are 327 Km/h and

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

296 km/h, respectively. This further supports the use of 321.8 km/h as a bounding value for Darlington Nuclear site.

The missile spectrum in Table 2.6-6 is extracted from Table 2 of U.S. NRC RG-1.76 Rev1 (Reference 2.6-13), Region 2 values, which correspond to a maximum wind speed of 321.8 km/h.

**Table 2.6-5: DNNP Site Characteristics for Design Basis Tornado (Reference 2.6-9)**

Parameter	Value
Maximum Pressure Drop	6.3 kPa
Maximum Rotational Speed	257.4 km/h
Maximum Translational Speed	64.4 km/h
Maximum Wind Speed	321.8 km/h
Radius of Maximum Rotational Speed	45.7 km/h
Rate of pressure Drop	2.5 kPa/s

**Table 2.6-6: DNNP Site Tornado Missiles Spectrum for Maximum Horizontal Speed (Reference 2.6-9)**

Missile Type	Dimensions	Mass	Horizontal Velocity ( $V_{mh}^{max}$ )	Vertical Velocity ( $0.67$ of $V_{mh}^{max}$ )
Schedule 40 Pipe	0.168 m dia x 4.58 m long	130 kg	34 m/s	22.8 m/s
Automobile 5 m x 2 m x 1.3 m	5 m x 2 m x 1.3 m	1810 kg	34 m/s	22.8 m/s
Solid Steel Sphere	2.54 cm dia	0.0669 kg	7 m/s	4.7 m/s

### Hurricanes

A tropical cyclone is a rapidly rotating storm system characterized by a low-pressure centre. Depending on the wind speed, it can be designated as hurricanes, tropical storms, or tropical depressions. Based on the information presented in Subsection 3.5.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4), there is a very low probability of a hurricane directly impacting the DNNP site, and it describes the probable maximum tropical cyclone as unlikely to yield gusts of more than 100 km/h which is lower than that of the design basis tornado. As such, wind hazard from a hurricane is not considered further.

Additionally, the 2009 NK054-REP-01210-00013 (Reference 2.6-4) states that a tropical storm such as Hazel, which occurred in 1954, would be the worst-case scenario from systems of tropical origin. During this storm, Toronto Pearson reported over 150 mm of rain in 2 days with sustained winds of 92 km/h for 2 hours and multiple hours with winds of over 70 km/h, per Subsection 3.5.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4). Precipitation caused from a tropical cyclone is covered in Subsection 2.5.3.5.

### **2.6.7 Waterspouts**

A tornado forming over water is a waterspout. The Site Evaluation on Nuclear Safety Considerations in the 2009 NK054-REP-01210-00008 (Reference 2.6-5), Section 3.2, states tornadoes over water or waterspouts generally leave no trace and are rarely reported. Additionally, the report states it is less likely for tornadoes to form over water than over land.

However, the report assumes an equal distribution of tornadoes and waterspouts for a given area and calculates the probability of a tornado at DNNP site. It then concludes that with such a frequency, tornadoes can be characterized as a rare, but non-negligible threat and a study of a design basis tornado was required in order to estimate the probability of occurrences on-site. The DNNP site characteristics for design basis tornado is described in Subsection 2.6.6.

### **2.6.8 Dust Storms and Sandstorms**

The assessment for the potential of dust storms or sandstorms was captured in the 2009 NK054-REP-01210-00013 (Reference 2.6-4) where Subsection 3.5.5 states a lack of evidence of these phenomena was identified from an extensive search through the available meteorological information relevant to southern Ontario. Hence, neither dust storms nor sandstorms were identified as potential hazards since the possibility of occurrence for these phenomena at the DNNP site is deemed to be highly unlikely.

### **2.6.9 Snow and Ice Load, Freezing Rain, and Ice Storm**

#### **Snow and Ice Load**

The average daily snowfall recorded at the nearest monitoring station to the Darlington Nuclear site is between 3 cm and 5 cm from December to March, per Section 2.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5). Similarly, the daily snowpack is typically recorded at the same location, and its highest point tends to occur in January, with a mean value of 8.6 cm.

Table 2.6-7 shows under Loading 1 the characteristic value of 2.2 kPa for snow and ice load for reactor designs considered for the DNNP site, per Subsection B.1.3, Table 3 of the 2012 N-REP-01200-10000 (Reference 2.6-9).

The 2019 DNGS hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1) used the 1975 NBCC design criteria for the snowpack of 2.1 kPa (Loading 2 in Table 2.6-7).

For the DNNP, Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8) assumed that similar snowfall conditions to the ones experienced in DNGS are expected to occur at DNNP due to their proximity. In 2022, a study was performed in NK054-REP-02730-00004 Winter PMP Validation (Reference 2.6-15) where a 50-year recurrence snow fall depth and maximum one-day late winter rain load nearby Oshawa are used to calculate the roof loading. The resulting loading is 1.8 kPa, as shown in Table 2.6-7, Loading 3. Furthermore, Loading 4 of 1.71 kPa in Table 2.6-7 represents the calculated DNNP snow load based of an NBCC 100-year recurrence, following the recommendation of CSA N291:19 (Reference 2.6-16) and employing a 50- to 100-year conversion multiplying factor of 1/0.82, as described in the 2022 NK054-REP-02730-00004 Winter PMP Validation (Reference 2.6-15), noting CSA N291:19 (Reference 2.6-16) does not require adding WPMP. The DNNP estimated snow loads and winter PMP values listed in Table 2.6-7 for 50-year recurrence or 100-year recurrence with or without WPMP are equal or lower than the Darlington Nuclear site characteristic value (Loading 1) of 2.2 kPa listed in Subsection B.1.3, Table 3 of the 2012 N-REP-01200-10000 (Reference 2.6-9).

**Table 2.6-7: Snow Loads and Winter PMP Values for DNGS and DNNP (Reference 2.6-15)**

Loading ID	Nuclear Site	Values	Compliance Notes
1	Darlington Nuclear	<b>2.2 kPa</b>	Characteristic value for reactor designs considered for the site (2010 PPE - Reference 2.6-9)
2	DNGS	<b>Snow: 2.1 kPa</b>	Meets the 1975 NBCC requirements (2019 SNGS - Reference 2.6-1)
3	DNNP (50-year recurrence)	Snow: 1.4 kPa + WPMP: 0.4 kPa <b>= Total: 1.8 kPa</b>	Meets 2015 NBCC requirements for 50-year recurrence snowpack, plus 50-year recurrence winter PMP near Oshawa (2022 DNNP - Reference 2.6-15)
4	DNNP (100-year equivalent recurrence)	Snow: (1.4/0.82) <b>= Total 1.71 kPa</b>	Meets 2015 NBCC requirements and CSA N291:19 requirements using a multiplying ASCE/SEI 7-10 factor of 1/0.82 to calculate the 100-year recurrence snowpack (2022 DNNP - Reference 2.6-15), noting N291:19 does not require adding WPMP.

#### Freezing Rain

With respect to freezing rain, Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8) indicates this item was considered for assessment as part of the safety analysis for DNNP. The hazards associated with freezing rain were also screened out for DNNP due to low consequence, as indicated in the 2019 hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1) and in the 2019 Site Preparation Nuclear Safety Licence Renewal Activity Report NK054-REP-01210-00108 (Reference 2.6-2).

#### Ice Storm

Ice storms present a potential hazard for the systems located outside the DNNP BWRX-300, as indicated in Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8). According to Subsection 4.5.5 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), a review of operating experiences indicates minor ice storms have not had an impact on other plants, but significant storms have caused losses of off-site power and switchyard failures. This event is described as an LOPP and is covered in Chapter 15, Subsection 15.5.3.2.4.

### **2.6.10 Lightning**

The assessment of lightning strikes is provided in Subsection 3.5.3 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) in the context of frequency of occurrence, where Table 3.5.10 provides estimates of the cloud-to-ground flashes for Toronto and Trenton, while Figure 3.5.8 displays graphically the Average Annual Flash Density in southern Ontario. Based on the data evaluated, the vicinity of the DNNP site will likely experience a frequency of 2 to 3 cloud-to-ground flashes per year per square kilometer. The 2020 DNNP lightning data collected and evaluated per NK054-CORR-00531-10533 (Reference 2.6-8) confirmed lightning occurrences are frequent in southern Ontario.

Subsection 4.5.7 of the 2019 Hazard Screening Assessment NK38-REP-03611-10043 (Reference 2.6-1) for DNGS summarizes the potential consequences of lightning occurrences as induced fires and electromagnetic compatibility issues affecting the functionality of electrical

systems. As shown in Table 4-3 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), the criterion assigned for lightning events is screening criterion QL-1 for DNGS, which is described as “an event of equal or lesser damage potential than similar events for which the plant has been designed.” This screening criterion is applicable to the DNNP site on the basis that adequate measures, such as fire barriers and qualification for electromagnetic compatibility, are incorporated in the BWRX-300 design, as described in Chapter 7, Section 7.1 and Section 7.3.

#### **2.6.11 Windborne Debris**

An analysis of windborne debris from various categories of high wind, also known as wind-propelled missiles, is assessed as part of the 2020 high wind PSA per NK054-CORR-00531-10533 (Reference 2.6-8). This assessment evaluated the impact of windborne debris on severe core damage and large release analysis. Tornado windborne missile hazard design basis is described in Table 2.6-6 in Subsection 2.6.6, Tornadoes and Hurricanes.

#### **2.6.12 Climate Change Impact**

As described in Subsection 2.5.4, the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.6-19) is developed with the objective of summarizing life cycle climate change considerations including relevant long-term monitoring that is described in Subsection 2.11.9.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**2.6.13 References**

- 2.6-1 NK38-REP-03611-10043 R003, 2019, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 2.6-2 NK054-REP-01210-00108 R000, 2019, "Site Preparation Nuclear Safety Licence Renewal Activity Report," Ontario Power Generation.
- 2.6-3 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.6-4 NK054-REP-01210-00013 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 4: Evaluation of Meteorological Events," Ontario Power Generation.
- 2.6-5 NK054-REP-01210-00008 R001, 2009, "Site Evaluation for OPG New Nuclear at Darlington - Nuclear Safety Considerations," Ontario Power Generation.
- 2.6-6 NK054-REP-01210-00016 R002, 2012, "Site Evaluation of the OPG New Nuclear at Darlington - Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.
- 2.6-7 NK38-SR-03500-10001 R005, 2018, "Darlington Safety Report, Part 1 and 2," Ontario Power Generation.
- 2.6-8 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation Licence (PRSL)," Ontario Power Generation.
- 2.6-9 N-REP-01200-10000 R003, 2010, "Use of Plant Parameters Envelope to Encompass the Reactor Designs Being Considered for the Darlington Site," Ontario Power Generation.
- 2.6-10 NK054-REP-01210-00078 R007, 2021, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 2.6-11 N-REP-NGD-IR-61100-0002, 1990, "NGD Meteorological Towers System Description and Operating Recommendations," Ontario Power Generation.
- 2.6-12 NK054-CORR-01210-1015770 R00, 2022, "Engineering Direction for Darlington Nuclear Project Design Basis Tornado Values," Ontario Power Generation.
- 2.6-13 U.S. NRC Regulatory Guide 1.76 Rev 1, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, July 2007.
- 2.6-14 NK054-REP-02730-00003, 2022, "Wind Gust Analysis," Ontario Power Generation
- 2.6-15 NK054-REP-02730-00004, 2022, "Winter PMP Validation," Ontario Power Generation
- 2.6-16 CSA N291:19, "Requirements for Nuclear Safety-related Structures," CSA Group.
- 2.6-17 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.
- 2.6-18 NK054-REP-02730-00002, 2022, "PMP Validation," Ontario Power Generation.
- 2.6-19 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation.

## 2.7 Geology, Seismology and Geotechnical Engineering

Section 2.7 covers the following DNNP site-specific information:

- Site Location and Description - Subsection 2.7.1  
Subsection 2.7.1 presents a general description of the site and identifies the study areas considered for the characterization of the site geological and geotechnical conditions
- Geological Characteristics - Subsection 2.7.2  
Subsection 2.7.2 contains the geological characteristics of the site including descriptions of the site physiography, surficial and bedrock geology, and offshore bathymetric contours and lakebed geology
- Geotechnical Characteristics - Subsection 2.7.3  
Subsection 2.7.3 describes the geotechnical and geological data collected at the site, presents subsurface soil and rock profiles and groundwater conditions, and provides an assessment of potential geotechnical hazards on structures
- Seismology Characteristics - Subsection 2.7.4  
Subsection 2.7.4 summarizes the seismological characteristics of the site including descriptions of the regional geology and tectonic history, hazard models, regional seismicity and seismic sources, ground motion characterization, methodologies used for the PSHA, and geological hazards that could potentially affect the site and the plant design.
- Geotechnical and Seismological Requirements and DNNP Site Parameters - Subsection 2.7.5  
Subsection 2.7.5 presents geotechnical and seismological parameters for the DNNP site including evaluation of bearing capacity and settlement, static and dynamic properties of rock, soil and engineered fill materials, geotechnical variability and uncertainty, Site Response Analysis (SRA), and groundwater level

The presented summary of geological, seismological, and geotechnical characteristics of the DNNP site and the surrounding region are based on:

- Site-specific characteristics from DNNP documents including the PSHA and the geological mapping of subsurface soil layers and bedrock, as well as relevant Darlington Nuclear site data.
- Available information developed during the DNNP site selection and preparation stages

In 2022 and 2023, several DNNP site-specific investigations and studies are completed as follows:

1. NK054-REP-01210-00175 R001, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project," Volumes 1 of 2 and 2 of 2 (Reference 2.7-39)
2. NK054-REP-10180-00001 R000, 2023 "Offshore Geotechnical Investigation," (Reference 2.7-40)
3. NK054-REP-03500.8-00001 R000, 2022, "Darlington New Nuclear Project - Site-Specific Probabilistic Seismic Hazard Assessment", (Reference 2.7-41)
4. NK054-REP-03500.8-00002 R000, 2022, "Darlington New Nuclear Project - Seismically-Induced Soil Liquefaction Assessment," (Reference 2.7-42)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

5. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," (Reference 2.7-38)

These investigations, assessments and analyses are used to validate and update DNNP-specific geological and geotechnical characteristics and parameters of subgrade materials, results of PSHA, potential of liquefaction underneath the BWRX-300 facility buildings, as well as Lake Ontario bathymetry and lakebed geology.

### **2.7.1 Site Location and Description**

The Darlington Nuclear site, where the DNNP BWRX-300 facility is to be built, is located about 65 km east of the City of Toronto on the north shore of Lake Ontario in the Municipality of Clarington, Region of Durham in Ontario, Canada. The DNNP site is located to the east of the existing DNGS as shown in Chapter 1, Appendix A, Figure A1.1-2. The site is at latitude 43° 53' north and longitude 78° 43' west, per the 2009 site geotechnical aspects evaluation NK054-REP-01210-00011 (Reference 2.7-1). (Refer to Section 2.1 for further information on the Darlington Nuclear site and the DNNP site description, layout, geography, and demography.)

The topography of the Darlington Nuclear site, shown in Figure 2.7.1-1, based on the Darlington Topographic Drawing NK054-DRAW-01210-00003 (Reference 2.7-26), indicates a gentle slope rising upward towards the east from an approximate elevation of 80 m to 88 m CGD, in a horizontal distance of about 400 m. Further east, the existing ground rises substantially to an elevation of about 100 m CGD near the east site boundary. The existing shoreline along the Darlington Nuclear site consists of a narrow beach with steep bluffs. Additional information about the Darlington Nuclear site topography is provided in Subsection 2.1.1.

The site is situated in an undulating to moderately rolling glacial till plain. However, the upper soils at the site are glaciolacustrine, indicating the site is in the Iroquois Plane. The previously irregular terrain was graded for the existing DNGS to an elevation of about 78 m CGD. For the DNNP, the terrain is planned to be graded to a grade elevation of 88 m CGD. The surface elevation for the DNNP site rises towards the north with a mean elevation of 100 m CGD just south of the Canadian National Railway tracks. To the north of the railway tracks, the ground is irregular ranging from 98 m to 106 m CGD. A higher ridge, starting from the shore just east of Raby Head, extends diagonally across the site in a northwesterly direction with levels of up to 15 m above the surrounding ground. Offshore from the site, the Lake Ontario bottom slopes away gradually reaching a depth of 6 m at about 425 m from shore and 14 m at approximately 1.2 km from shore. Offshore bathymetry is discussed in Subsection 2.7.2.4.

### **2.7.2 Geological Characteristics**

Summaries based on the information in the 2009 DNNP Site Geotechnical Aspects Evaluation NK054-REP-01210-00011 (Reference 2.7-1), the 2013 DNNP Geotechnical Data Report NK054-REP-01210-00098 (Reference 2.7-29), the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40), and the 1989 DNGS Preoperational Summary Report No. 89575 (Reference 2.7-2) are presented in:

- Subsection 2.7.2.1 - Surficial Geology
- Subsection 2.7.2.2 - Site Physiography
- Subsection 2.7.2.3 - Bedrock Geology
- Subsection 2.7.2.4 - Offshore Bathymetric Contours / Lakebed Geology

These summaries furnish a framework within which the geological characteristics of the DNNP site and the surrounding region are described.

### **2.7.2.1 Surficial Geology**

The regional surficial geology, for an area within an approximately 50 km radius from the DNNP site, is shown in Figure 2.7.2-1, as replicated from the 2009 NK054-REP-01210-00011 (Reference 2.7-1).

For the surficial geology, there are three general physiographic regions:

- The Oak Ridges Moraine on the north side of the regional study area
- The South Slope in the middle
- The Iroquois Plain, a wide belt along Lake Ontario in the south

#### **The Oak Ridges Moraine Physiographic Region**

The Oak Ridges Moraine is a significant geologic/hydrogeologic feature specific to southern Ontario. The moraine is a major source of groundwater recharge, and many creeks and rivers are derived from groundwater discharge from the moraine. It was formed by regional glaciation, the advance and recession of several ice sheets and the subsequent melting of the glaciers. The moraine marks the boundary between the Lake Simcoe ice lobe advancing from the north and the Lake Ontario ice lobe advancing from the south. It is a ridge of high land separating drainage northward to Lake Simcoe and southward to Lake Ontario.

The moraine consists of interbedded layers of glacial till, sand and gravel. The moraine has a distinctive hummocky terrain with knobs and kettles. The southern flank of the moraine is covered by the Halton Till, a silty to silt-clay till.

#### **The South Slope Physiographic Region**

The South Slope fills the area between the moraine and the Iroquois Plain. It consists of gentle to steep slopes but is more uniform compared to the irregular terrain of the moraine. It contains a number of drumlins which point to the southwest, indicating the general direction of glacier movement.

#### **The Iroquois Plain Physiographic Region**

The Iroquois Plain, an 8 to 12 km wide plain, lies between the former shoreline of Lake Iroquois and present-day Lake Ontario. Shoreline deposits and glaciolacustrine sediments are found in this area overlying the glacial tills. The shoreline deposits include sand and gravel bars and beach terraces as well as some deltas from former rivers and creeks flowing into Lake Iroquois. The lacustrine deposits, consisting of silts and clays overlying till are found further from the former shoreline. In the area of the site, the Iroquois Plain contains drumlins with a southeast orientation indicating the northwest glacial advance.

### **2.7.2.2 Site Physiography**

The DNNP site is generally covered by upper and lower till deposits, per the 2009 NK054-REP-01210-00011 (Reference 2.7-1), as described in the following paragraphs.

The surface till in the DNNP area is similar to the Newmarket Till, a sandy silt to silt till. An earlier dense, to very dense, sandy silt to hard silty clay till overlies the bedrock. Bounded between the upper and lower tills are deposits of water-bearing sand or sand and gravel.

Earlier deposits of lacustrine varved silt and clay and stratified fine to medium sand overlie the upper till at lower elevations near the DNNP BWRX-300 location, as described in the 2013 NK054-REP-01210-00098 (Reference 2.7-29). These surficial lacustrine deposits consist of varved silt and clay and fine to medium sand of variable thickness, per the 2013 NK054-REP-01210-00098 (Reference 2.7-29).

Fill material of variable composition is present at the ground surface over portions of the DNNP site, as described in the 2013 NK054-REP-01210-00098 (Reference 2.7-29). The fill consists of a mixture of clay, silt, sand, and gravel.

Overburden thickness varies significantly from the north to the south. Overburden thickness in the Oak Ridges Moraine is approximately 200 m reducing in thickness towards the south with about 10 m of overburden at Lake Ontario.

### **2.7.2.3 Bedrock Geology**

The bedrock is completely covered by Quaternary deposits and bedrock outcrops are found only in local quarries, as described in the 2009 NK054-REP-01210-00011 (Reference 2.7-1). The bedrock surface, from east to west, consists of the Simcoe Group overlain by the younger Blue Mountain (formerly the Whitby Formation) and Georgian Bay Formations. The Simcoe Group consists of the Gull River, Bobcaygeon, Verulam and Lindsay Formations (from deep to shallow). The dip of the bedrock formations is approximately 0.5 percent to the southwest.

The Blue Mountain Formation is a shale formation. The lower 2 m to 3 m includes what was formerly known as the Whitby Formation, a black, petroliferous calcareous shale which tends to weather grey on exposure. The shale is fissile and fossiliferous. The Lindsay Formation is a grey argillaceous limestone with a full formation thickness of approximately 67 m.

The Verulam, Bobcaygeon and Gull River Formations lie below the Lindsay Formation. They are shale and limestone formations. The Shadow Lake Formation, a sandstone and shale formation, lies unconformably on the Precambrian Basement, as explained in the 2009 NK054-REP-01210-00011 (Reference 2.7-1).

Based on the described bedrock geology, the bedrock at the site of the DNNP is mainly the Lindsay Formation overlying the Verulam and Bobcaygeon and Gull River Formations. The upper few meters of bedrock are shaley limestone and shale of the Blue Mountain Formation that overlies the Simcoe Group, as detailed in the 2013 NK054-REP-01210-00098 (Reference 2.7-29).

### **2.7.2.4 Offshore Bathymetric Contours / Lakebed Geology**

The bathymetric contours of the lakebed along Lake Ontario shoreline of the Darlington Nuclear site are provided in the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40). This investigation was conducted to characterize the lakebed and sub-bottom materials and profile the depth to bedrock. The offshore geophysics methods used were:

- Seismic reflection
- Sub-bottom profiling
- Electrical resistivity tomography
- Multi-beam echosounder
- Side scan sonar
- Magnetometer

The most prominent feature of the lakebed topography reported in the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40) is a crescent shaped ridge and peninsula of shallower depths which wraps from the northeast to the west of the surveyed area. The shape of this ridge creates a deeper “bay” in the central west part of the surveyed area; to the southeast the lakebed drops off into deeper water, as shown in Figure 2.7.2-3 and Figure

2.7.2-4. The results are aligned with previous studies of the offshore bathymetry and lakebed surface geology, per the 1989 Report No. 89575 (Reference 2.7-2), as depicted in Figure 2.7.2-5 and Figure 2.7.2-6.

### **2.7.3 Geotechnical Characteristics**

Subsection 2.7.3 includes the following information related to the geotechnical characteristics of the DNNP site:

- Subsection 2.7.3.1 describes available geotechnical and geological data collected for the DNNP site
- Subsection 2.7.3.2 presents subsurface stratigraphic soil and rock profiles and groundwater conditions at the DNNP site
- Subsection 2.7.3.3 provides an assessment of potential geotechnical hazards on the DNNP structures

#### **2.7.3.1 Geotechnical Information Collected at the DNNP Site**

Multiple geotechnical investigations have been completed for the DNNP site. The data compiled in the investigations described in this subsection are used in determining the static and dynamic subgrade properties of the DNNP site presented in Subsection 2.7.5.

##### ***2.7.3.1.1 CH2MHILL (2007, 2008) Study***

The investigation was performed by CH2MHILL in late 2007 and early 2008 and included installing monitoring wells in 11 borings. The results of this study are presented in two reports, the 2009 DNNP Geotechnical Aspects Site Evaluation NK054-REP-01210-00011 (Reference 2.7-1) and the 2009 DNNP Geological and Hydrogeological Environment NK054-REP-07730-00005 (Reference 2.7-30). These boreholes covered an area larger than the boundary of the DNNP site. The locations of the monitoring wells and the corresponding borehole numbers (DN) within the area planned for the construction of the DNNP in the CH2MHILL study, are marked with red circles in Figure 2.7.3.1-1.

##### ***2.7.3.1.2 AMEC (2012) Study***

Three vertical boreholes completed within the DNNP area by AMEC in the 2012 DNNP Geologic and Geophysical Evaluation NK054-REF-01210-0418696 (Reference 2.7-28) are used to obtain subsurface information to the depth of the Precambrian Basement rock. The results of this study are presented in the 2012 NK054-REF-01210-0418696 (Reference 2.7-28). The locations of these deep borings are shown in Figure 2.7.3.1-2. The boreholes included: AMC-01 to a depth of 231.6 m, AMC-02 to a depth of 239.6 m, and AMC-03alt to a depth of 239.6 m. This study provides detailed boring logs, downhole geophysical measurements including televiewer data, surface geophysical measurements, and laboratory testing results. The data compiled in this study was mainly used to characterize the bedrock units. The geotechnical data provided in this AMEC study for the soil units are limited.

##### ***2.7.3.1.3 EXP Service INC. (2013) Study***

In the 2013 DNNP Geotechnical Data Report NK054-REP-01210-00098 (Reference 2.7-29), eight sampled boreholes were drilled at locations within the DNNP area as shown in Figure 2.7.3.1-3. The drilled boreholes were advanced to various depths between 34 m to 85 m below the surface. The geotechnical data include detailed stratigraphic information, results of in-situ Standard Penetration Tests (SPTs) with calibrated hammers, and data from laboratory testing of soil and rock samples. Subsurface cross-section diagrams developed as part of the EXP study are presented in Figure 2.7.3.1-4 and Figure 2.7.3.1-5.

#### **2.7.3.1.4 WSP GOLDER (2022) Phase 1 Geotechnical Investigation Report**

In the 2022 Geotechnical Investigations NK054-REP-01210-00175 (Reference 2.7-39), extensive drilling was conducted at locations within the DNNP area to determine engineering properties of soil and rock, with specific focus on the first BWRX-300 location as shown in Figure 2.7.3.1-6. The stratigraphic units identified for the DNNP site and corresponding description are listed in Table 2.7-1. The site investigation followed the guidelines of NEDO-33914-A (Reference 2.7-27), Section 3.1, to ensure an adequate characterization of the subsurface conditions that meet additional requirements specific to the BWRX-300 design as a deeply embedded Small Modular Reactor (SMR).

Sampling was conducted in conjunction with in-situ SPTs performed with calibrated automatic hammers and data from laboratory testing of soil and rock samples are outlined in Section 4 and Section 5 of the 2022 Phase-1 investigations report (Reference 2.7-39).

The methodology for the in-situ and laboratory test are outlined in Volume 1 – Factual Geotechnical Data Report of NK054-REP-01210-00175 (Reference 2.7-39). The types of tests conducted include:

- Soil chemical analysis for the following constituents:
  - Soil pH of soil for corrosion
  - Water-soluble sulfate
  - Chloride in water
  - Sulfate in water for concrete
- Soil resistivity analysis
- Vane shear tests (cohesive soils)
- Pressuremeter testing (soil), dilatometer testing (rock), piezocone soundings (soil), soil resistivity, packer testing (rock), over-coring stress testing (rock)
- Uniaxial Compression Stress (UCS) testing (rock)
- Triaxial compression stress testing (soil)
- Constant stress direct shear creep testing on rock joints
- Swell testing (rock)

#### **2.7.3.2 Subsurface Stratigraphic Profile**

##### **2.7.3.2.1 Profiles for the DNNP Site (2022)**

The stratigraphy for the DNNP site soil and bedrock units listed in Table 2.7-1 is developed based on the work performed in the 2022 Geotechnical Investigations NK054-REP-01210-00175 (Reference 2.7-39).

Details of the in-situ stratigraphic layers average and range of thicknesses are provided in Table 2.7-2 for the soil units and in Table 2.7-3 for the rock units. The interpreted soil and rock stratigraphy are presented in east-west oriented and north-south oriented cross-sections in Figure 2.7.3.2-1 and Figure 2.7.3.2-2, respectively. Further details for subsurface soil and bedrock profiles are described in the following paragraphs.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Unit 1 – Topsoil/Fill

The uppermost layer is topsoil and/or fill consisting of either poorly graded sand with gravel or sandy lean clay. Unit 1 was encountered at ground surface at all boreholes drilled for the 2022 Phase 1 Geotechnical Report NK054-REP-01210-00175 (Reference 2.7-39). This layer has an average thickness of 1.59 m ranging from 0.53 m in borehole 27 to 3.53 m in borehole 67.

Units 2a and 2b – Surficial Glaciolacustrine Deposits

Two glaciolacustrine deposits are found below the upper topsoil and fill. The upper deposits (Unit 2a) are encountered below the topsoil/fill layer. Unit 2a consists of silt, clay, fine to coarse sand and trace to some subrounded to subangular gravel. The lower deposits (Unit 2b) consist of silt with some clay, fine to coarse sand and subrounded to angular, fine to coarse gravel.

In some boreholes, Units 2a and 2b were observed to be interlayered. The combined thickness of Units 2a and 2b averages 1.74 m, and is ranging from zero in borehole 6, to 6.1 m in borehole 5.

Unit 3 - Upper Till

Deposits of silty sand with gravel to sandy lean clay with gravel are encountered below Units 2a and 2b. Unit 3 is described as a till layer generally consisting of a heterogeneous mixture of dense to very dense gravel, boulders, and cobbles in a matrix of silty sand. This deposit consists of silt, clay, fine to coarse sand and subrounded to subangular to angular, fine to coarse gravel. Unit 3 ranges in thickness from zero in borehole 4 to 13.49 m in borehole 17, with an average thickness of 7.35 m.

Units 4a and 4b – Intermediate Glaciolacustrine Deposits

Two distinct glaciolacustrine deposits are founded below Unit 3. The upper deposit, Unit 4a consists of silt, clay, fine to coarse sand and subrounded to angular, fine to coarse gravel. Boulders and cobbles are also present within Unit 4a. Below Unit 4a is Unit 4b which consists of silt, clay, fine to coarse sand and trace to some subrounded to angular gravel.

In some boreholes, Units 4a and 4b were observed to be interlayered. The combined thickness of units 4a and 4b averages 11.3 m, and ranges between zero in borehole 11SB to 17.7 m in borehole 27.

Unit 5 – Lower Till

Below the intermediate glaciolacustrine deposits (Units 4a and 4b), a deposit of very dense silt and sand to hard lean clay (Unit 5) is encountered. Unit 5 is described as a lower till layer generally consisting of a heterogeneous mixture of gravel, boulder, and cobbles in a matrix of silt sand and silty clay. This deposit consists of silt, clay, fine to coarse sand, and subrounded to angular, fine to coarse gravel. It has an average thickness of 3.57 m, ranging from zero in borehole 16 to 6.63 m in borehole 15.

Unit 6a – Blue Mountain Formation Bedrock

The top of the bedrock is at an average elevation of 64.20 CGD, ranging from 62.72 m CGD in borehole 6 to 65.80 m CGD in borehole 70.

Below Unit 5, is a moderately weathered to fresh, very thinly to medium bedded, fine grained, faintly porous, slightly to moderately reactive to hydrogen chloride, weak to strong shale with thin, limestone interbeds. Unit 6a has an average thickness of 2.98 m, ranging from 1.38 m in borehole 73 to 5.87 m in borehole 30.

**Unit 6b – Lindsay Formation Bedrock**

Below Unit 6a is a slightly weathered to fresh, very thinly to medium bedded, fine to medium grained, faintly porous, slightly to moderately reactive to hydrogen chloride, weak to medium strong to very strong limestone with shale interbeds, Unit 6b has an average thickness of 61.36 m, ranging from 60.61 m in borehole 16 to 61.93 m in borehole 65.

**Unit 6c – Verulam Formation Bedrock**

Below Unit 6b is a fresh very thinly to medium bedded, grey, fine to medium grained, faintly porous, moderately reactive to hydrogen chloride, medium strong to very strong limestone with shale interbeds. Full thickness of Unit 6c was not tested.

**Table 2.7-1: Stratigraphic Units for the DNNP Site**

Unit No.	Description
1	Topsoil / Fill
2a	Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay
2b	Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt
3	Upper Till
4a	Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt
4b	Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay
5	Lower Till
6a	Blue Mountain Formation Bedrock
6b	Lindsay Formation Bedrock
6c	Verulam Formation Bedrock

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.7-2: In-situ Soil Units Stratigraphy under the Power Block (Reference 2.7-39)**

Layer	Layer Thickness (m)							
	Reactor Building <sup>1</sup>		Power Block <sup>2</sup>		BWRX-300 Protected Area <sup>3</sup>		BWRX-300 Study Area <sup>4</sup>	
	Average	Range	Average	Range	Average	Range	Average	Range
Unit 1	1.25	0.61 – 2.13	1.81	0.61 – 3.28	1.77	0.61 – 3.53	1.59	0.53 – 3.53
Unit 2a Unit 2b	1.73	0.61 – 3.81	2.32	0.00 – 6.09	2.35	0.00 – 6.09	1.74	0.00 – 6.10
Unit 3	6.24	1.07 – 8.87	6.01	0.00 – 9.06	6.26	0.00 – 13.47	7.35	0.00 – 13.49
Unit 4a, Unit 4b	9.32	0.00 – 14.48	9.78	0.00 – 14.32	9.07	0.00 – 14.54	11.30	0.00 – 17.70
Unit 5	2.29	1.36 – 2.98	3.78	1.36 – 6.63	3.25	0.86 – 6.63	3.57	0.00 – 6.63

Notes:

1. Includes borings BH 9, BH 10, BH 11, BH 11S, BH 11 SB, BH 12, BH 14 (Reference 2.7-39)

2. Includes borings BH 2, BH 4, BH 5, BH 9, BH 10, BH 11, BH 11S, BH 11SB, BH 12, BH 13, BH 14, BH 18, BH 19, BH 67, BH 68, BH 71, BH 73, BH 78 (Reference 2.7-39)

3. Includes borings BH 2, BH 4, BH 5, BH 6, BH 8, BH 7, BH 9, BH 10, BH 11, BH 11S, BH 11SB, BH 12, BH 13, BH 14, BH 15, BH 16, BH 17, BH 18, BH 19, BH 20, BH 66, BH 66S, BH 66SB, BH 67, BH 68, BH 71, BH 73, BH 77, BH 78 (Reference 2.7-39)

4. All boreholes considered in the study area in (Reference 2.7-39)



**Table 2.7-3: Rock Units Stratigraphy (Reference 2.7-39)**

	Layer Thickness or Depth (m)	
	Average	Range
<b>Elevation</b> Top of Bedrock	64.20 (CGD)	62.72 (BH 6) - 65.80 (BH 70)
<b>Thickness</b> Unit 6a - Blue Mountain Formation	2.98	1.38 (BH 73) - 5.87 (BH 30)
<b>Thickness</b> Unit 6b - Lindsay Formation	61.36	60.61 (BH 16) - 61.93 (BH 65)

Notes:

1. Full thickness of the Verulam Formation (Unit 6c) was not tested (Reference 2.7-39)
2. Lindsay formation thickness determined from small sample ~ (15%) of boreholes which extended fully through the formation (Reference 2.7-39)

#### **2.7.3.2.2 Planned As-Built Soil Profile**

Stratigraphic Units 1 and 2 are generally loose, have liquefaction potential (Subsection 2.7.4.7.6), and are not suitable for supporting the heavy foundations of the power block buildings. As a result, during site development, these soil layers will be excavated and replaced with compacted engineered fill.

Consequently, the as-built conditions at the site after construction of the BWRX-300 facility are anticipated to include compacted engineered fill from about elevation ranging between 80 m to 82 m CGD to the final grade at elevation 88 m CGD. The excavated soil from this site may be used as compacted engineered fill material if it meets the engineered fill gradation requirements outlined in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

#### **2.7.3.2.3 Bedrock Profile**

The bedrock profile was developed based on readily available top-of-rock information from boreholes drilled for the geotechnical study in the 2022 NK054-REP-01210-00175 (Reference 2.7-39). Data between boreholes have been interpolated.

The top of the bedrock surface undulates relatively locally and slopes gently to the south from an elevation of 67 m CGD near the northern extent of the site to an elevation of 64 m CGD. This bedrock surface is consistent with the mapped sub-horizontal dip of the Paleozoic sequence observed within the vicinity of the project area.

Subsurface rock conditions may vary between and beyond the borehole/drillhole locations. The interpreted stratigraphy is therefore a simplification of the subsurface bedrock contacts. Variations in the stratigraphic boundaries between boreholes/drillholes will exist and are to be expected. Table 2.7-3 presents the top of bedrock elevation and bedrock thicknesses.

The BWRX-300 deeply embedded RB is anticipated to extend through the Blue Mountain Formation (Unit 6a) and be founded in the Lindsay Formation (Unit 6b) at 52.93 m CGD. The top of the Blue Mountain Formation near the BWRX-300 RB is anticipated to be at about 64 m CGD based on the depth to bedrock at BH 10, BH11 and BH 12 (refer to Figure 2.7.3.1-6), as explained

in the 2022 Power Block geotechnical investigations NK054-REP-01210-00175 (Reference 2.7-39).

#### **2.7.3.2.4 Groundwater Conditions**

Based on the information provided in the 2022 DNNP Phase 1 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39), the following three groundwater flow patterns are identified:

- The Unit 3 water table (shallow groundwater), shown in Figure 2.7.3.2-3
- Unit 4a groundwater flow in the integrated deposits, shown in Figure 2.7.3.2-4
- Unit 5 groundwater in the interglacial deposits located above the bedrock, shown in Figure 2.7.3.2-5
- Units 6a-6b groundwater in bedrock, shown in Figure 2.7.3.2-6

The groundwater flow interpretations in these figures (Figures 2.7.3.2-3 to 2.7.3.2-6) are based on a monitoring well-network with only a few months of monitoring data. The actual long-term interpretation may change. The contours are based on data from the new monitoring wells installed within the investigation area, which are limited in aerial extent, and have not been considered with contemporary groundwater elevation data from the pre-existing monitoring well-network at the site.

As shown on the figures, the groundwater flow direction in the upper and lower till (Units 3 and 5, respectively) is inferred to be toward the southwest and, in the intermediate glaciolacustrine deposits (Unit 4a) and shallow bedrock (Units 6a and 6b), to be toward the south-southeast.

Regional groundwater flow and flow at the DNNP site generally follows topography from higher elevations in the north towards the south, per the 2009 DNNP Geological and Hydrogeological Environment NK054-REP-07730-00005 (Reference 2.7-30). In general, this flow is driven by recharge from rainfall and snowmelt infiltration across the area and at higher elevations along the Oak Ridges Moraine north of the DNNP site with discharge, ultimately, to Lake Ontario to the south. The shallow groundwater system at the DNNP site deviates from this flow pattern near surface water conveyances and local recharge areas. Interpreted regional groundwater flow patterns documented in the 2009 report NK054-REP-01210-00011 (Reference 2.7-1) are shown in Figures 2.7.3.2-7, 2.7.3.2-8 and 2.7.3.2-9 for shallow water table, interglacial deposits, and shallow bedrock groundwater, respectively.

The hydro-stratigraphic units at the DNNP site follow the soil and geologic units. The upper till (Unit 3) forms an aquitard or confining layer at the site which restricts downward groundwater flow from the upper fill and glaciolacustrine materials. The interglacial deposits (Units 4a-4b) are the most significant hydrogeologic unit at the site since they extend across the site and to the recharge areas north of the site, as described in the 2009 NK054-REP-07730-00005 (Reference 2.7-30). There may be significant groundwater flow in the interglacial deposits due to the higher gradient and higher permeability of the materials. The lower till (Unit 5) beneath the interglacial deposits is also considered an aquitard with low permeability. Although flow in the upper till is downward due to under-draining by the interglacial deposits, there may be an upward component of flow through the lower till in some areas from the underlying upper bedrock aquifer, per the 2009 NK054-REP-07730-00005 (Reference 2.7-30).

The upper bedrock is likely fractured and weathered with higher secondary permeability and transmissivity compared to the intact bedrock. Flow in the upper bedrock is expected to be enhanced in areas where the lower till is absent, and the upper bedrock is in direct contact with the more permeable interglacial deposits. The lower bedrock at the DNNP site generally has low

permeability and does transmit much groundwater. The groundwater conditions in the deeper bedrock formations below the Lindsay Formation have not been considered for study.

### **2.7.3.3 Evaluation of Geological Hazards on DNNP Structures**

Subsection 2.7.3.3 provides an assessment of geological hazards that could impact the DNNP structures.

#### **2.7.3.3.1 Karst Cavities**

No evidence of significant karst cavities was encountered in the 2022 geotechnical boreholes (Reference 2.7-39). Some zones of lost core were encountered in the boreholes generally within 40 m of the ground surface and ranging from 5 cm to 66 cm in height, but no noticeable drop in the drilling rods was noted and therefore these are thought to be associated with zones of weathered and fragmented rock that had been washed out by the core drilling.

In addition, the previous geophysical reports associated with this site indicate the absence of anomalies in the rock that could indicate the presence of depressions or voids that may be indicative of large karst or faults. There is good seismic coverage with nine seismic refraction lines being executed at the site.

Review of the previous surface geophysical data as well as the numerous boreholes drilled in 2021 to 2022 for the power block, and the associated data (Reference 2.7-39) confirm the absence of karst features at this site.

#### **2.7.3.3.2 Ground Frost**

The conventional approach for protection of building foundations against frost action is to locate base of foundations and/or the base of grade beams (supported on deep foundation) at a depth at least equal to the depth of frost penetration. A minimum frost penetration depth of 1.3 m is therefore recommended, in accordance with OPSD 3090.101 (Foundation Frost Penetration Depths for Southern Ontario), as per the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39). Partial or complete frost protection may also be achieved by using rigid polystyrene insulation.

Frost heaving may occur in fine grained soils where ice lenses occur when moisture is drawn to freezing horizons. Based on the existing site subsurface conditions, shallow silty fine sand and silt deposits below surficial granular fill are generally expected to be moderately to highly frost susceptible to heaving under freezing conditions. Therefore, adequate frost cover of 1.3 m depth is required for all foundations exposed to frost conditions.

#### **2.7.3.3.3 Bearing Failure (Collapse)**

The 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) evaluated the bearing capacities for the RB foundation and resulting bearing capacities for the Turbine Building (TB), Control Building (CB), Radwaste Building (RWB), and Reactor Auxiliary Bay foundations surrounding the deeply embedded RB using data reported in the 2022 geotechnical site investigations (Reference 2.7-39). The anticipated bearing pressure and bearing capacity for each building in the power block is summarized and discussed in Subsection 2.7.5.1.

#### **2.7.3.3.4 Stability of Foundation**

The 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) provides the anticipated maximum uniform and differential settlements of the RB, TB, CB, RWB, and Reactor Auxiliary Bay foundations. The anticipated bearing pressure and associated settlements are summarized and discussed in Subsection 2.7.5.1.

#### ***2.7.3.3.5 Stability Of Subgrade Surrounding the Reactor Building***

A stability analysis was performed following the guidelines of NEDO-33914-A (Reference 2.7-27), Section 4.0, using the finite element software PLAXIS (Bentley) to perform advanced non-linear Soil-Structure Interaction (SSI) numerical modeling. In addition to the stability analysis, the potential for instability of the potentially unstable blocks or wedges surrounding the RB deep excavation were performed using UnWedge (RocScience), a 3D stability analysis and visualization program. The stability analysis is discussed further in Subsection 2.7.5.1 and all the analyses are detailed in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

#### ***2.7.3.3.6 Transitional Ground Heave and Settlement***

As part of site grading and development, there will be unloading and transitional ground heave resulting from excavation of the upper soft-to-loose soil layers of Units 1 and 2 at the site. Additionally, some of the heave will be offset by settlement, which will occur on completion of backfilling. Depending on the net change in the overall effective stress profile, net ground heave is expected to occur due to reduction in the finished ground level compared to existing levels.

During the process of unloading and re-loading, stratigraphic Units 3, 4 and 5 are expected to react quickly to the changes in the ground stresses with minimal lag. Hence, long-term consolidation or heave is not expected to occur. Rather only transitional elastic rebound, and compression are expected to occur, as documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

It is anticipated that there will be about 10 mm of heave from offloading due to excavation and some nominal heave/settlement after the completion of fill placement. There may be some ongoing creep settlement from the fill placement; however, ground movements will be small and the impact on structures founded on or in the overburden soils will be insignificant, as described in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

#### ***2.7.3.3.7 Stability of Natural Slopes***

The structures located within the power block are level at finished grade and over 100 m away from the shoreline. The structures are expected to be founded on or in either engineered fill, very dense native Unit 3 soil or deeply embedded in strong to very strong bedrock. Therefore, slope instability will not be threat to these power block structures. However, the natural shoreline is prone to erosion, especially the steep bluffs to the east of the power block area. Erosion of the shoreline has the potential to pose a hazard eventually, through gradual reduction of the ground pressure, if allowed to progress over long periods. This is discussed in the 2022 NK054-REP-03500.8-00002 Darlington New Nuclear Project - Seismically-Induced Soil Liquefaction Assessment (Reference: 2.7-42). Prevention of erosion is to be achieved through the establishment of engineered shoreline protection. The steep bluffs as a slope do not pose a hazard to the first BWRX-300 unit planned, and design of subsequent units will mitigate the hazard as required.

#### ***2.7.3.3.8 Stability of Cut and Fill Slopes***

The existing ground to the east of the existing DNGS will be excavated to form a large level area for the DNNP and its associated structures. For preliminary design purposes, cut slopes into the competent interglacial/till deposits will be at a general inclination of 1V:3H (18.4°). The excavated soils will be partially stored at the north-east part of the site. The fill slopes will be designed to ensure stability.

#### **2.7.3.3.9 *Stability of Dikes and Dams***

No dams are currently present or planned for the DNNP. No dikes are currently present or planned on DNNP, and lake infilling is no longer planned for the project.

#### **2.7.4 Seismology Characteristics**

Subsection 2.7.4 summarizes findings of past seismic hazard investigations as well as of the 2022 site-specific PSHA (Reference 2.7-41) that were performed for the DNNP and DNGS site.

Subsection 2.7.4 includes:

- Subsection 2.7.4.1 - provides background seismological information and data collected since 1997
- Subsection 2.7.4.2 - describes the regional geological structure and tectonic history of the Darlington Nuclear site
- Subsection 2.7.4.3 - presents information on the seismicity of the region surrounding the site and the development of earthquake catalogue
- Subsection 2.7.4.4 - describes the seismic hazard model containing regional and local sources
- Subsection 2.7.4.5 - describes aspects related to ground motion characterization
- Subsection 2.7.4.6 - discusses the PSHA methodology and the results for the DNNP site
- Subsection 2.7.4.7 - describes protentional geological and seismological aspects at the DNNP site

##### **2.7.4.1 Background and Data Collection**

In 2009, the Darlington Nuclear site was evaluated for suitability for the DNNP. A PSHA was performed, per the 2009 NK054-REP-01210-00014 (Reference 2.7-4) in accordance with:

- CNSC Regulatory Document RD-346 Site Evaluation for New Nuclear Power Plants (Reference 2.7-5), which is superseded by CNSC's REGDOC 1.1.1 Site Evaluation and Site Preparation for New Reactor Facilities (Reference 2.7-6)
- IAEA NS-R-3 (Reference 2.7-7), which is superseded by SSR-1 (Reference 2.7-8)

The 2009 PSHA (Reference 2.7-4) details assembly of the geological, geophysical, and seismological data collection for the region, near region and vicinity of the DNNP site. The approach adopted utilized the 1997 study (Reference 2.7-3) as a starting point. The database assembled for that study was updated, and the effects of the updates of regulatory requirements in CNSC RD-346 (Reference 2.7-5) and IAEA NS-R-3 (Reference 2.7-7) were evaluated, and changes were incorporated. The 2009 PSHA was thereafter revised three times: in 2011 in NK38-REP-03611-10041 R000 (Reference 2.7-9), in compliance with CSA Standard N289.2 (Reference 2.7-31); in 2019 in NK38-REP-03611-10041 R002 (Reference 2.7-10), and in 2021 in NK38-REP-03611-10041 R003 (Reference 2.7-11), with minor changes to address CNSC comments not previously incorporated. The PSHA updates in both the 2019 NK38-REP-03611-10041 R002 (Reference 2.7-10) and the 2021 NK38-REP-03611-10041 R003 (Reference 2.7-11) include:

- Updates to the Earthquake Catalogue
- Updates to the Maximum Magnitude Assessment
- Updates to Earthquake Occurrence Rates

- Application of the Next Generation Attenuation -East Ground Motion Model

In 2022, a DNNP site-specific PSHA (Reference 2.7-41) was conducted in accordance with the requirements of CNSC REGDOC 2.5.2 and CSA N289 series, as well as with the BWRX-300 SMR specific design requirements listed in NEDO 33914-A (Reference 2.7-27). In addition, the 2022 PSHA study (Reference 2.7-41) used the 2001 NUREG/CR-6728 (Reference 2.7-20) to develop site-specific ground motions considering local site conditions.

## **2.7.4.2 Regional Geological Structure and Tectonic History**

### ***2.7.4.2.1 Regional Geological Structure Stratigraphy***

The Darlington Nuclear site lies within the western Lake Ontario region in the tectonically stable interior of the North American continent, which is characterized by low rates of historical seismicity, as described in the 1994 EPRI TR-102261-V1 (Reference 2.7-12). The region is underlain by middle Proterozoic (about 900 to 1600 million years ago) Grenville basement rock and overlying Paleozoic (about 250 to 570 million years ago) shallow-water sedimentary strata.

The Grenville Province formed in response to several phases of compression and metamorphism. The “Grenville Front” and “Grenville Front Tectonic zone”, shown in Figure 2.7.4.2-1, is the contact between the Grenville Province to the east and the continental Eastern Granite-Rhyolite provinces to the west. Rocks of the Central Gneiss Belt are between the “Grenville Front Tectonic Zone” and the Central Metasedimentary Belt Boundary Zone. The Central Metasedimentary Belt Boundary Zone underlies the western end of Lake Ontario, and the Central Metasedimentary Belt underlies the rest of Lake Ontario and the site study region. The Central Metasedimentary Belt is an intensely faulted and folded zone formed less than 1,300 million years ago. The southeastern portion of the Central Metasedimentary Belt consists of slightly younger rock. The Grenville orogeny (mountain-building episode) is widely attributed to a continental collision; however, deformation occurred in several episodes of extension and compression.

The Grenville Province’s crustal structure is characterized by north-northeast-striking, relatively shallow east-southeast-dipping ductile thrust faults that developed at mid- to lower-crustal depths during the middle Proterozoic Grenville orogeny. Prominent north-northeast-trending geophysical anomalies associated with exposed Grenville structures extend southward beyond the Canadian Shield and beneath the unconformable lower Paleozoic cover rocks. Regional geologic maps (e.g., Ontario Geological Survey, 1991) indicate that the overlying Paleozoic rocks are, with few exceptions, relatively flat-lying and laterally continuous, indicating that no large-scale, major faulting has occurred in the region since they were deposited.

The notable exception to the lack of regional-scale faulting in southern Ontario and Quebec occurs within the St. Lawrence rift system, as described in the 1966 Canadian Journal of Earth Sciences, Volume 3, No. 5 (Reference 2.7-13), which is a remnant of the late Proterozoic/early Paleozoic Iapetus passive margin, as described in the 1996 published article of R.L. Wheeler (Reference 2.7-14). The St. Lawrence rift system comprises abundant large-scale normal faults displacing lower Paleozoic strata and underlying Grenville basement on the order of many hundreds of meters along the Ottawa, Champlain, St. Lawrence, and Saguenay River valleys (Reference 2.7-13). These extensional faults generally cut discordantly across Grenville-aged structures instead of reactivating them. Mesoscopic-scale faulting of the lower Paleozoic strata, with fault displacements ranging from less than a meter to several tens of meters, has been recognized locally throughout much of the Lake Ontario region outside of the St. Lawrence rift system. The St. Lawrence rift system is associated with zones of elevated and persistent seismicity, per Slemmons, D.B., et al. in 1991 (Reference 2.7-15).

Worldwide, the seismic potential of a stable continental region varies according to the degree of crustal extension that it experienced in the geologic past, and to a lesser extent, the age of the

crust, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12). Three types of crust are identified in eastern North America, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12):

- Unrifted - the craton and the Appalachian fold belt
- Failed intracontinental rifts—the Ottawa and Saguenay aulacogens and the Reelfoot rift complex
- Rifted passive continental margin—the Atlantic passive margin produced by the present opening of the Atlantic Ocean in the late Mesozoic, and a relic passive margin produced by lapetan rifting in the late Proterozoic/early Paleozoic

The north-northeast-trending faults along the Champlain and St. Lawrence River valleys, once attributed to a two-sided, failed intracontinental rift, are now recognized as part of the southeast-facing lapetan margin, per R. L. Wheeler, in 1996 (Reference 2.7-14). The present-day Atlantic passive margin comprises transitional crust (continental-oceanic) and the extended and faulted inboard continental shelf.

Evidence of lapetan rifting of the craton adjacent to the northern Appalachians is recorded within the St. Lawrence rift system (Reference 2.7-13) in the form of rift-related extensional structures, sediments, and magmatic/volcanic products that developed along the ancient continental margin. The rift structures include zones of echelon faults parallel to the ancient margin, possible fracture zones transverse to the ancient margin, and two well-defined aulacogens (failed rifts)—the Ottawa and Saguenay grabens.

The Appalachian orogen lies approximately 400 km east of the Darlington Nuclear site. Northern Appalachian orogenic events occurred from Ordovician to Permian time and consisted of several distinct tectonic episodes. As discussed in the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41), the key structural elements that mark the boundaries of the various crustal provinces (e.g., the western limit of Mesozoic extensional structures) are used to define regional seismic source zones that are characterized by similar crustal properties (for an example of one boundary interpretation, refer to Figure 2.7.4.2-2).

#### **2.7.4.2.2 *Neotectonics Setting***

The geologically most recent evidence for major tectonic activity in the region is Alleghanian (late Permian) thrust faults formed in the Appalachian foreland basin and late Triassic to late Jurassic normal faults along the Atlantic margin related to continental rifting and the subsequent opening of the Atlantic Ocean, per the 2009 NK054-REP-01210-00014 DNNP PSHA (Reference 2.7-4). However, historical seismicity along the St. Lawrence rift system, in the Charleston, South Carolina, area, and in other concentrated zones; local geologic evidence of Cenozoic reactivation of faults; evidence of seismically-induced liquefaction in susceptible sands and silts; and geologic and geodetic data indicative of regional and local crustal deformation suggest continuing neotectonic activity, albeit at much lower rates than during the last episode of major tectonic deformation.

Slemmons, D.B., et al. in 1991 (Reference 2.7-15) have reported that most large historical and instrumental earthquakes in eastern Canada have occurred near Paleozoic or younger rift zones. This is similar to stable continental region earthquakes worldwide, as described in the 1994 EPRI TR-102261-V1 (Reference 2.7-12). The early Paleozoic St. Lawrence rift system, which is delineated by a persistent pattern of seismicity, is the postulated source of numerous large, historical earthquakes in southeastern Canada, per Slemmons, D.B., et al. in 1991 (Reference 2.7-15). Seismicity along this rift system appears to be concentrated in a number of well-defined clusters, including the Ottawa River, Charlevoix, and lower St. Lawrence River seismic zones, which are all separated by relatively aseismic regions.

Equivocal evidence for neotectonism, per Thomas, R.L., et al. in 1993 (Reference 2.7-16), has been found in the Lake Ontario region, and there are difficulties in distinguishing between deformation related to glacial processes and that related to deep-seated tectonic processes.

East-northeast/west-southwest-trending lakebed features in the Rochester basin of Lake Ontario and the Hamilton-Presqu'île fault zone, along with some of the features observed in western Lake Ontario, have been proposed by Thomas, R.L., et al. in 1993 (Reference 2.7-16), as neotectonic evidence for the southwest continuation of the St. Lawrence rift system through Lakes Ontario and Erie.

The postulated northwestern boundary of the late Proterozoic/early Paleozoic lapetan rifted margin tectonic province lies approximately 80 km east of the site, per Wheeler, R.L. in 1995 (Reference 2.7-17). There also is deep seismic evidence suggesting that the western boundary of the lapetan margin may lie farther to the west, along the Central Metasedimentary Belt Boundary Zone of the Grenville province as described by Milkereit, B., et al. in 1992 (Reference 2.7-18). These alternative boundaries are considered in defining regional seismic source zones (for an example of one boundary interpretation, refer to Figure 2.7.4.2-2).

The rate of historical seismic activity in the Grenville Province west of the lapetan rifted margin is low and appears typical of stable cratonic crust, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12). In general, seismic activity and the geologic conditions most associated with earthquake activity in the stable continental region of Central and Eastern North America increase towards the east, away from the Precambrian central craton and towards the rifted passive continental margin.

#### **2.7.4.3 Seismicity**

Characterization of the seismicity of the region surrounding the DNNP site forms an essential part of the assessment of the seismic hazard. The primary means of characterization of seismicity is the use of the earthquake catalogue to assess earthquake occurrence rates and maximum magnitudes for earthquake sources.

In the Darlington Nuclear site PSHA studies, presented in the 2019 NK38-REP-03611-10041 R002 (Reference 2.7-10) and the 2021 NK38-REP-03611-10041 R003 (Reference 2.7-11), the 2012 NUREG-2115 (Reference 2.7-21) earthquake catalogue was updated to include independent earthquakes from the end of 2008 through 20 May 2019. The earthquake catalogue was again updated in the 2022 NK054-REP-03500.8-00001 DNNP PSHA (Reference 2.7-41) to extend the duration of the catalogue to the end of December 2021 using the:

1. National Earthquake Database of Canada
2. U.S. Geological Survey earthquake catalogue
3. Weston Observatory earthquake catalogue

The 2012 NUREG-2115 (Reference 2.7-21) contains data collected through mid-2009. Expected moment magnitudes were determined for the added earthquakes as described in 2022 PSHA NK054-REP-03500.8-00001 (Reference 2.7-41).

Figure 2.7.4.3-1 depicts the spatial distribution of earthquakes in the updated de-clustered catalogue exclusively in the time window between 2008 and December 31, 2021, as described in the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference-2.7-41).

The maximum magnitude ( $M_{\max}$ ) distributions for the distributed seismicity sources (seismicity source zones) were obtained using the project earthquake catalogue and the methodology developed in NUREG-2115 (Reference 2.7-21). The project earthquake catalogue was also used to obtain updated earthquake recurrence assessments for the seismic sources



#### **2.7.4.4 Seismic Source Characterization**

The seismic source zonation model used in the 2022 DNNP PSHA NK54-REP-03500.8-00001 (Reference 2.7-41) is that presented in the 2021 Darlington PSHA NK38-REP-03611-10041 R003 (Reference 2.7-11) with the exception of updates to the 2020 Geological Survey of Canada historical seismicity zonation (H model) based on Adams, et al. (Reference 2.7-19). The seismic source characterization model comprises regions of distributed seismicity and local sources representing identified geological/geophysical features. An overview of the information in the 2022 DNNP PSHA (Reference 2.7-41) with respect to the regional and local seismic sources is summarized in the following paragraphs.

##### ***2.7.4.4.1 Regional Source Zones***

Three alternative approaches to regional seismic zonation are used to represent the sources of distributed seismicity throughout the study region. Figure 2.7.4.4-1 presents the logic tree structure used in the 2022 DNNP PSHA (Reference 2.7-41) as well as previous PSHA studies in the Darlington Nuclear site area, representing the epistemic uncertainty in regional seismic source zonation. The three alternative approaches are used to define the source zonation for distributed seismicity sources as follows:

1. The favored approach (weight 0.8) was to define source zones on the basis of seismotectonic evaluations. Epistemic uncertainty in defining the boundaries between these seismotectonic sources led to the set of alternative zonations.
2. An alternative approach (weight 0.1) was to use the historical seismicity zonation developed by the Geological Survey of Canada as part of the Canadian National Earthquakes Hazards Program (Adams, J., et al., 2019) (Reference 2.7-19) These regional Seismicity Zones are shown in Figure 2.7.4.4-2.
3. The third alternative was to use a zoneless model (weight of 0.1) in which seismicity parameters were defined for individual cells comprising 1 degree longitude by 1 degree latitude within the study region shown in Figure 2.7.4.4-3.

##### ***2.7.4.4.2 Local Source Zones***

There are six potential local seismic source zones that are defined based on their identified geological/geophysical features, per the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41). These six source zones are: Clarendon-Linden Fault System, Georgian Bay Linear Zone, Hamilton-Presqu'île Fault, Mississauga Magnetic Domain, Niagara-Pickering Linear Zone, and Wilson-Port Hope Magnetic Lineament. These sources act as potential concentrators of seismic activity and are critically assessed for their seismogenic potential. The locations of these sources have been extracted from the 2022 DNNP PSHA (Reference 2.7-41) and are depicted in Figure 2.7.4.4-4.

#### **2.7.4.5 Ground Motion Characterization**

Ground motion models are needed to calculate the effects at the site of earthquakes occurring in the characterized seismic sources. Two aspects are considered as follows:

1. Estimation of the amplitude of ground motions as a function of earthquake size and the source-to-site distance that is provided by ground motion models
2. Assessment of the effect of the local site conditions on the generic hard rock ground motions by results of site response analyses performed in a manner that achieves hazard-consistent ground motions at the site surface

In active tectonic environments, ground motion models are often developed from the analysis of recorded strong motion data. The seismic hazard was computed using the Pacific Earthquake Engineering Research Center model documented in the 2018 PEER Report No. 2018/08 by Goulet, C., et al. (Reference 2.7-22). The model is the most comprehensive ground motion model available for Central and Eastern United States (CEUS) Seismic Source Characterization.

#### **2.7.4.6 PSHA Results for the DNNP Site**

The 2022 PSHA study in NK054-REP-03500.8-00001 (Reference 2.7-41) presents the seismic hazard characterization for the deeply embedded BWRX-300 RB at the DNNP site. The study meets the requirements and follows the guidance of CNSC REGDOC-2.5.2 (Reference 2.7-32), CSA N289 Series (Reference 2.7-31, 2.7-32, and 2.7-33), and the Licensing Topical Report NEDO-33914-A (Reference 2.7-27).

The PSHA presented in the 1994 EPRI TR-102261-V1 (Reference 2.7-12) and the 2021 Darlington Risk Assessment (Reference 2.7-11), developed Uniform Hazard Response Spectra (UHRS) for rock outcropping motions at the anticipated level of the foundation of the DNGS plant at the top of the Paleozoic bedrock strata. The DNGS foundation level is not at the same elevation as the foundation of the BWRX-300 deeply embedded RB.

The seismic hazard model used in the 2022 DNNP PSHA (Reference 2.7-41) is based on the seismic hazard model employed in the 2021 Darlington Nuclear site PSHA (Reference 2.7-11) and is updated using new data and information. Differences between the two seismic hazard models, overall, are minor and include:

- Recalculated earthquake recurrence parameters, such as rates, maximum magnitude ( $M_{\max}$ ), and spatial distributions, using the updated earthquake catalogue
- Slight increase in the probability that the Wilson-Port Hope local source is associated with small magnitude earthquake, resulting from additional earthquakes being recorded in the region (this produces a minor increase in the overall probability of activity for this source)
- The source zone geometry for zonation based on historical seismicity is updated to be consistent with the 6<sup>th</sup> Generation of seismic hazard maps of Canada, H2 model for source zonation, per Adams, J., et al. (Reference 2.7-19).

The approach to site-specific hazard differs between the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41) and the 2021 Darlington site PSHA NK38-REP-03611-10041 (Reference 2.7-11). In the 2021 Darlington site PSHA (Reference 2.7-11) site-specific hazard results were obtained solely for the reactor basemat elevation for the existing DNGS using the two options for application of the EPRI 2006 Cumulative Absolute Velocity (CAV) model specified in USNRC (2012a) (Reference 2.7-35):

- Option 1 specified computing the hazard integrating from a minimum magnitude of M 5 (M 4 was used) but only applying the CAV filter to the contributions from magnitudes less than M 5.5
- Option 2 specified computing the hazard integrating from a minimum magnitude of M 5 without applying the CAV filter. Deterministic site amplification functions from reference rock were computed using a site profile truncated at the reactor foundation elevation.

Epistemic uncertainty in site amplification scaling reference rock motions to foundation level motions was incorporated into the CAV calculations but aleatory variability in amplification was not included. Vertical motions were obtained by applying mean V/H ratios to the horizontal UHRS.

#### **2.7.4.6.1 Site Response Analysis**

Site-specific hazard in the 2022 DNNP PSHA (Reference 2.7-41) is computed only using USNRC (2012a) (Reference 2.7-35) Option 2, integration of hazard from M 5 without applying the EPRI (2006) (Reference 2.7-36) CAV filter. Site amplification was computed using NUREG/CR-6728 Approach 3, per McGuire et al., in 2001 (Reference 2.7-20). This approach develops the SRA in which probabilistic site amplification functions defining both median amplification and aleatory variability in amplification were convolved with the reference rock hazard to produce site-specific hazard at the target elevations. Epistemic uncertainty in site amplification was modeled.

The site response model was extended to finish grade to represent anticipated as-built site conditions with reactor basemat elevation for the planned BWRX-300 is located approximately 12 m below the top of rock at the DNNP site while the reactor foundation levels at the existing DNGS site are at or near top of rock. Minimum epistemic uncertainty in site amplification was applied in both studies, with the updated value used for the DNNP study being 50 percent larger than the value used in the 2021 NK38-REP-03611-10041 (Reference 2.7-11). Seismic hazard results for vertical motions were computed by convolving probabilistic V/H ratios with the horizontal hazard rather than applying mean V/H ratios.

Per guidance of NEDO 33914-A (Reference 2.7-27), Section 5.2.2, the site-specific hazard is defined for the following three horizons at:

- The RB foundation bottom elevation 52.93 m CGD
- The soil/rock interface elevation 64 m CGD
- The finished grade elevation 88 m CGD

There are only slight differences between the reference rock and site-specific hazard curves at the RB base and soil/rock interface as presented in the 2022 DNNP PSHA report (Reference 2.7-41). The horizontal mean hazard curves were interpolated to obtain UHRS for an Annual Frequency of Exceedance (AFE) of 1E-2, 1E-3, 1E-4, 1E-5, 1E-6, and 1E-7 for the RB base, soil / rock interface, and finished grade elevations, respectively. The results of the UHRS curves at the horizontal and vertical of the three targeted horizons are provided in Figure 2.7.4.6-1 through Figure 2.7.4.6-8.

Seismic hazard results were produced in the 2022 DNNP PSHA report NK054-REP-03500.8-00001 (Reference 2.7-41) for Design Basis Earthquake (DBE) seismic inputs to design and Beyond Design Basis Earthquake (BDBE) seismic inputs for the evaluations of the Design Extension Conditions (DEC) as per REGDOC-2.5.2 (Reference 2.7-32) and to the Checking Level Earthquake as per CSA N289.1:18 (Reference 2.7-33). Section 9.2 of NK054-REP-03500.8-00001 (Reference 2.7-41) describes the development of DBE and BDBE ground motion response spectra. Figure 2.7.4.6-9 through Figure 2.7.4.6-11 compare the DBE and BDBE horizontal ground motion spectra with the corresponding UHRS with 1E-4 and 1E-5 AFE for the three elevations mentioned above.

Subsection 2.7.5.3.5.1 presents the DBE and BDBE response spectra that define the amplitude and frequency content of the DBE and BDBE ground motion. The DBE horizontal ground motion spectra meet the minimum earthquake requirement by enveloping the CSA N289.3 minimum spectrum as shown in Figure 2.7.4.6-12 and Figure 2.7.4.6-13.

Subsection 2.7.5.2.5.2 presents the hazard-consistent, strain-compatible dynamic soil properties used as input for the seismic response analysis and design of BWRX-300 RB that were also developed for both the DBE and BDBE levels of motion using the results of the site response analyses.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Augmentations were applied to the DBE and BDBE RB base motions, as described in Section 9.4 of NK054-REP-03500.8-00001 (Reference 2.7-41), to produce foundation input response spectra which meet the requirements of the 2010 USNRC DC/COL ISG-017 (Reference 2.7-37) for hazard consistency of foundation input response spectra for SSI analyses following guidance of NEDO 33914-A, Section 5.3.4.1. Finally, sets of recorded ground motions were recommended for use as seed motion in developing time histories for seismic analyses.

Table 2.7-4 identifies the figures which present UHRS based on the mean hazard results, reproduced from the 2022 PSHA NK054-REP-03500-.8-00001 (Reference 2.7-41).

**Table 2.7-4: Figures Presenting UHRS Based on Mean Hazard Results.**

<b>Elevation (m CGD)</b>	<b>Orientation</b>	<b>Figure</b>
52.93	Horizontal	2.7.4.6-1
52.93	Vertical	2.7.4.6-2
64	Horizontal	2.7.4.6-3
64	Vertical	2.7.4.6-4
88	Horizontal	2.7.4.6-5
88	Vertical	2.7.4.6-6
Reference Rock	Horizontal	2.7.4.6-7
Reference Rock	Vertical	2.7.4.6-8
52.93	Horizontal DBE and BDBE	2.7.6.4-9
64	Horizontal DBE and BDBE	2.7.6.4-10
88	Horizontal DBE and BEBE	2.7.6.4-11

#### **2.7.4.7 Potential Seismically Related Hazards**

Several geological hazards and seismicity-related phenomena that could potentially affect the suitability of the DNNP site and the plant design are evaluated.

##### **2.7.4.7.1 *Volcanism***

A methodology for initial investigation of volcanism suggests evaluating within a 150 km radius of the site, per the 2009 DNNP Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.7-23). The methodology states that if there is no evidence of Cenozoic era (i.e., within the last 65 million years), volcanic rocks or volcanism in the region, no further investigations are required. Geological Map 1860a from Natural Resources Canada in the 2009 DNNP NK054-REP-01210-00012 (Reference 2.7-23) does not identify Cenozoic era formations within 150 km of the site. Hence, volcanism at the DNNP site is considered an improbable hazard with no associated seismic activity.

##### **2.7.4.7.2 *Tsunami***

Tsunamis are long period gravity waves generated in oceans or lakes by seismic disturbances or landslides resulting in a sudden displacement of the water surface. The resulting wave energy spreads across the ocean or lake at high speed. Tsunami occurrences in Canada are rare, with the Pacific Coast at greatest risk due to the higher occurrence rate of earthquake and landslide activity. The magnitude 7.2 Grand Banks earthquake of 1929 produced tsunami effects on the Burin Peninsula of Newfoundland. The Great Lakes are on the edge of the Canadian Shield, a

geologically stable, mid-continental region where the rate of occurrence of earthquakes is about one tenth of that at tectonic plate boundaries.

The Lake Ontario shorelines are not generally susceptible to shore slope failure or landslide. Review of U.S. National Geophysical Data Center Lake Ontario bathymetry gave no evidence of submarine landslides or other surface disturbance in the post-glacial period, per the 2009 NK054-REP-01210-00012 (Reference 2.7-23). Around the perimeter of Lake Ontario, "Quaternary sediments are relatively thin or absent, and bedrock exposures are common, possibly reflecting the effects of sub-glacial erosion and subsequent abrasion by lacustrine waves and currents."

The Natural Hazards Database at the U.S. National Geophysical Data Center reports one 1755 "tsunami run-up event" in Lake Ontario, though this appears to have been a seiche-like event. The event, for a location about 50 km northwest of Rochester, N.Y. is coded as "an event that only caused a seiche or disturbance in an inland river", source "unknown." "In Lake Ontario the water repeatedly rose in an unusual way to the height of about 1.5 m, no shock is mentioned. Exact latitude and longitude are unknown."

In the absence of tsunami reports in Lake Ontario and the lack of shoreline or lakebed evidence of tsunami initiators, tsunamis are considered improbable events with no associated flood hazard potential at the site.

#### **2.7.4.7.3 Seiches**

Storm surge and seiche effects in Lake Ontario resulting for various scenario storms were considered in the 2009 NK054-REP-01210-00012 (Reference 2.7-23). The maximum storm-induced surge and seiche at the Darlington shore is 0.75 m. The 1755 event where 1.5 m high seiche-like oscillations in Lake Ontario were reported may not have been seismically-induced as no shock is mentioned. A review of historical earthquake records in the 2009 DNNP PSHA NK054-REP-01210-00014 DNNP PSHA (Reference 2.7-4) identified an event on January 9th, 1847, in Grafton Harbour where with "Lake Ontario calm under a north wind, suddenly the lake level descended, exposing the lakebed for upwards of about 107 m". In moments it recoiled, rushing towards the shore in one unbroken wave about 1.2 m above normal. This wave accompanied by a heavy noise crashed over the wharf and washed inland about 91 m. This happened about 8 or 9 times, each with "diminishing force." The editor of the Cobourg Star reminded his readers that something similar had occurred in Cobourg and Port Hope in 1845. An apparently related report described "some commotion" at Rice Lake about 19 km north of Grafton Harbour, during which the 0.46 m of ice on Rice Lake began "to undulate". Eventually the ice burst with "a noise like thunder" and chunks in the center of the lake were tossed into a pile about 3.1 m high. These reports do not mention ground shaking, although noise is mentioned.

Based on the historical evidence, seiche events have occurred in Lake Ontario; therefore, shoreline protection at DNNP is considered in the design as discussed in Subsection 2.4.2.

#### **2.7.4.7.4 Dams and Landslides**

There are no human-made water retaining structures within the Darlington Creek watershed or other site vicinity watersheds, as described in the 2009 Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.7-23). Hence, there are no flooding hazards associated with seismically-induced failure of human-made water retaining structures. Additionally, the flooding threat due to seismically-induced landslide at the site is minimal, per the 2009 NK054-REP-01210-00012 (Reference 2.7-23). These conclusions are validated in Section 5.6 of the 2022 DNNP Site Evaluation Update Summary Report NK054-REP-01210-00142 (Reference 2.7-43).

#### **2.7.4.7.5 Surface Faulting**

At present, there is no known evidence of larger, pre-historic earthquakes that have resulted in surface fault rupture because such earthquakes have not occurred, or the evidence for surface rupture or coseismic damage is not preserved, or the studies needed to identify past large earthquakes is insufficient to recognize these events.

Given the relatively stable geological setting of the region surrounding the Darlington site, the recency of the post-glacial landscape that might preserve past large earthquake effects, it is expected that evidence for large earthquakes if they have occurred, would be difficult to identify. The 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39) confirms the absence of historical evidence for surface rupture within the Darlington site, including any absence of faults within the boreholes as logged.

The onshore and offshore boreholes and mapping of the DNGS excavations did not indicate offsets in the stratigraphic units, shear zones, or deep depressions in the bedrock surface, hence no near surface faulting has occurred in the bedrock at the site, as described in the 1981 DNGS Geotechnical Mapping of Bedrock Excavation NK38-02004P (Reference 2.7-24). There is no evidence of post-glacial fault-related scarps in the overburden or of solution-weathered cavities in the bedrock, as reported in the 1977 DNGS Geology and Seismicity - Hydro Geotechnical Engineering Dept. Report 77110 (Reference 2.7-25).

The stratigraphic continuity of the upper Paleozoic bedrock in the site vicinity conformed to the regional dip of about 5 m/km to the south. Minor changes in thickness and position of marker units were evident, but the differences were well within the limits of variation expected for sedimentary rock formations in southern Ontario. No vertical dislocation or displacement was evident in the upper Paleozoic bedrock formations, indicating that faulting has not propagated through the sedimentary rock strata from the Precambrian basement rock.

Mapping of marker units in the DNGS intake and discharge tunnels that extend over 1 km south of the site showed continuity consistent with the regional dip. Jointing in the rock is tight and water ingress is insignificant.

Regional geologic maps, e.g., Ontario Geological Survey, 1991, indicate that the Paleozoic rocks are, with few exceptions, relatively flat-lying and laterally continuous, indicating that no large-scale, major faulting has occurred in the region since they were deposited.

The 2022 DNNP Geotechnical Investigation NK054-REP-01210-00175 (Reference 2.7-39) reaffirmed the conclusions from the 2009 investigations and it is concluded that there is no evidence of surface faulting in the overburden or bedrock at the site or site vicinity.

#### **2.7.4.7.6 Liquefaction Potential of Foundations**

The RB foundation is to be founded on sound limestone bedrock. Foundations of other structures are to be founded on dense to very dense till deposits, and/or engineered fill. As such, the liquefaction potential of foundations will be low.

The 2022 DNNP Liquefaction Assessment Report NK054-REP-03500.8-00002 (Reference 2.7-42) assessed seismically-induced liquefaction hazards of foundation soils for the DNNP to support the Licence to Construct (LTC) application. The assessment considered the latest seismic hazard values reported in the 2022 DNNP PSHA NK054-REP-03500.8-00001 DNNP PSHA (Reference 2.7-41). The detailed liquefaction assessment of foundation soils was performed for the structures No. 1 to No. 6, namely, (1) RB, (2) TB, (3) RWB, (4) CB, (5) Reactor Auxiliary Bay, and (6) Independent Spent Fuel Storage Installation (ISFSI), as labelled in Figure 2.7.4.7-1 and Figure 2.7.4.7-2. In addition, for the potential Emergency Mitigating Equipment (EME) Access Routes at the site, all boreholes within the project boundary were evaluated for liquefaction potential.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

The following conclusions were derived from the results of the liquefaction assessment (Reference 2.7-42).

- For the DBE event, foundation soil liquefaction is not expected for the structures within the power block including the RB, TB, RWB, CB, and Reactor Auxiliary Bay, based on available soil data and the plan for the power block area to be over-excavated approximately to elevation 81 m CGD and then backfilled to plant grade elevation of 88 m CGD. For the foundation soil below the structures No.2 to No.5, the estimated seismically-induced settlement is typically less than 5 mm with a maximum of 8 mm, and the seismically-induced lateral displacement is expected to be up to 28 mm under DBE event
- For the DBE event, soil in the vicinity of the ISFSI structure (Structure No. 6) is expected to experience liquefaction, particularly in the surficial glaciolacustrine deposit (Unit 2 from the expected finished grade at elevation 88 m down to about 5 m depth). The estimated seismically-induced settlement is up to 154 mm and the lateral spreading displacement is up to 1.67 m.
- For the BDBE event, foundation soil liquefaction is not expected for the following structures:
  - RB (Structure No. 1), founded directly on bedrock
  - TB (Structure No. 2)
  - RWB (Structure No. 3)
- For the BDBE event, liquefaction potential exists at only one data point (isolated and limited extent of zones) for foundation soils in the vicinity of the following structures:
  - CB (Structure No. 4) - The liquefaction data point is at about elevation 69.1 m CGD, about 18.9 m depth from the finished grade.
  - Reactor Auxiliary Bay (Structure No. 5) - The liquefaction data point is at about elevation 69.9 m CGD, about 18.1 m depth from the finished grade.
- For the BDBE event, the foundation soil of the structures No.2 and No.5 in the power block area is calculated to have typically less than 17 mm and up to 27 mm of seismically-induced settlement, and the displacement due to lateral spreading that is calculated to be typically less than about 0.05 m and up to about 0.09 m displacement, as per the detailed liquefaction assessment of the available geotechnical data.
- For the BDBE event, significant liquefaction and seismically-induced deformation is expected in the vicinity of the proposed location for the ISFSI structure (Structure No.6).
- For the EME access routes, liquefaction susceptibility and screening assessment was performed considering all boreholes (forty-eight in total) at the site except for those within the power block area. Figure 2.7.4.7-1 and Figure 2.7.4.7-2 show the locations of the boreholes which are susceptible to liquefaction for the DBE and BDBE events respectively.

In Section 7.2 of the 2022 DNNP Geotechnical Investigation Report (Reference 2.7-39), it is indicated the upper clayey, sandy, and silty deposits (i.e., Units 2a and 2b) are potentially liquefiable during the 10,000-year design earthquake event. However, approximately 8 m of soil will be removed from beneath the power block and replaced by engineered fill. Excavating the aforementioned soil units by the specified 8 m will mitigate the potential for liquefaction. It is therefore concluded that the soil under the power block is considered non-liquefiable under the 10,000-year design earthquake for the RB, TB, RWB, CB, and the Reactor Auxiliary Bay.



### **2.7.5 Geotechnical and Seismological Parameters**

Subsection 2.7.5 describes the site-specific information used for developing the geotechnical and seismological parameters for the in-situ site conditions prior to construction of and the anticipated as-built conditions after the construction of the BWRX-300 facility. The in-situ conditions are characterized based on the information described in Subsection 2.7.3, including the results reported in the 2022 NK054-REP-01210-00175 Phase I Geotechnical Investigation Report (Reference 2.7-39) and the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

Subsection 2.7.5 is divided into the following subsections:

- Subsection 2.7.5.1: Assessment of As-Built Conditions at the DNNP Site, including a description of the over-excavation and fill replacement, evaluation of bearing capacity and time-dependent deformation for the proposed foundations, and evaluation of the anticipated earth pressure on structures.
- Subsection 2.7.5.2: Geotechnical and Seismological Site Properties, including subgrade stratigraphic profiles, static and dynamic properties of rock and soil; and groundwater level
- Subsection 2.7.5.3: Geotechnical Variability and Uncertainty, including potential sampling bias, inherent variability of samples and possible measurements errors consideration, including the main source of epistemic and aleatory uncertainties

#### **2.7.5.1 Assessment of As-Built Conditions at DNNP Site**

The site geotechnical investigations, presented in Subsection 2.7.3, are used to characterize the stratigraphy of subsurface materials at the area of the DNNP site where the first BWRX-300 unit is to be constructed. The data collected from the 2022 geophysical investigations NK054-REP-01210-00175 (Reference 2.7-39) provide comprehensive understanding of the subsurface soil and the deep bedrock conditions at the site.

The DNNP site subsurface soil and rock profiles are presented in Subsection 2.7.3.2. The DNNP site consists of approximately 25 m of soil deposits overlaying bedrock. Both the soil and bedrock materials are characterized as flat laying to slightly dipping toward the south. The top and surficial soil deposits may not have the required capacity to support the near surface mounted foundations of the BWRX-300 RWB, TB, CB and Reactor Auxiliary Bay (refer to Chapter 1, Figure A1.1-2, Figure A1.4-1 and Figure A1.5-1 for site and BWRX-300 Unit 1 layouts). Bearing capacity and settlement confirmatory calculations were performed, as part of the 2022 geotechnical work NK054-REP-01210-00175 (Reference 2.7-39) and the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), considering approximate dimensions, bearing pressure demands and stratigraphy of the soil materials under the RWB, TB, and CB and the Reactor Auxiliary Bay foundations.

The results of the geotechnical investigations that are reported in the 2012 NK054-REF-01210-0418696 (Reference 2.7-28), the 2013 NK054-REP-01210-00098 (Reference 2.7-29), and the 2022 geotechnical investigations and tests (Reference 2.7-39) do not indicate the presence of rock cavities, voids, large open fractures, significant eroded zones, shear zones, or joint configurations that would have a potential for causing rock instability and thus jeopardizing the integrity or the safety functions of the deeply embedded BWRX-300 RB.

##### ***2.7.5.1.1 Over-excavation and Fill Replacement***

The range of SPT blow count numbers (as low as 6) and laboratory tests results indicate that the topsoil and fill materials may contain organic clays and be soft or very loose sands, which is not suitable for supporting the near surface mounted foundations of RWB, TB, CB, and Reactor

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Auxiliary Bay. As described in Subsection 2.7.3.1, beneath the topsoil and fill materials, two layers of surficial lacustrine soil materials that differ in clay content and plasticity were identified:

- The layer at the top (Unit 2a) consists of sandy lean clay to lean clay soil with soft to very stiff consistency
- The layer below (Unit 2b) consists of cohesionless silty gravel to silty sand materials, with compactness varying from very loose to very dense

The SPT blow counts taken for the two surficial lacustrine soil layers (Units 2a and 2b) show low values indicating that these materials may not be suitable for supporting the RWB, TB, CB, and Reactor Auxiliary Bay foundations and may liquefy during a DBE level event. The results of field and laboratory tests performed for the upper till (unit 3), intermediate glaciolacustrine (Units 4a and 4b), and lower till (Unit 5) indicate dense and stiff materials surrounding the deeply embedded RB that have no potential for liquefaction during a DBE event and are suitable for supporting the foundations of the RWB, TB, CB, and other power block structures.

As a result, site preparation for construction of the BWRX-300 SMR is anticipated to include excavation at the power block area of the weaker surficial soils to an elevation between 80 m and 82 m CGD. The excavated surface soils will be replaced with engineered fill to bring the site grade back to elevation 88 m CGD. The dense upper till, intermediate glaciolacustrine and lower till soils below elevations 80 m to 82 m CGD would remain in place. The BWRX-300 RB would then be constructed in a vertical right cylinder shaft excavation that extends to a depth of about 35.2 m or elevation 52.8 m CGD. At this depth, the bottom of deeply embedded BWRX-300 RB is anticipated to extend through the compacted or engineered fill and in-situ soils and into the underlying bedrock.

The RWB, TB, CB, and other power block structures surrounding the deeply embedded RB are anticipated to be supported by shallow foundations on the engineered fill.

Information detailed in the 2021 licensing topical report on BWRX 300 Advanced Civil Construction and Design Approach, NEDO-33914-A (Reference 2.7-27) describes the approach to be used for monitoring the effects of excavation and construction on the properties of subsurface materials; specifically in its Subsection 3.4 Field Instrumentation Plan, and Section 4.0 Foundation Interface Analysis.

#### ***2.7.5.1.2 Bearing Capacity Evaluation for Proposed Foundations***

##### ***2.7.5.1.2.1 Shallow Foundation***

As documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), based on engineering assessment, conventional spread and strip footings located in the power block area which are founded on engineered fill can be designed using ultimate bearing capacities ( $q_u$ ):

- 1.0 m wide with depths of 1.3 to 2.5 m: 1857 to 3642 kPa
- 2.0 m wide with depths of 1.3 to 2.5 m: 1854 to 3493 kPa
- 3.0 m wide with depths of 1.3 to 2.5 m: 1834 to 3509 kPa
- 4.0 m wide with depths of 1.3 to 2.5 m: 1854 to 3422 kPa
- 5.0 m wide with depths of 1.3 to 2.5 m: 1891 to 3393 kPa

Raft foundations can be used for heavily loaded structures where conventional spread or strip footings are not adequate to support. Raft foundation founded on engineered fill can be designed for the following ultimate bearing capacities ( $q_u$ ):

- 68 x 70 m TB raft foundation with depths of 1.3 to 2.5 m: 5672 to 6917 kPa
- 30 x 48 m RWB raft foundation with depths of 1.3 to 2.5 m: 3986 to 4978 kPa
- A factor of safety of 3.0 is recommended to be used for the service limit state, and a resistance factor of 0.5 is recommended to calculate the ultimate limit state

#### ***2.7.5.1.2.2 Reactor Building Deeply Embedded Foundation***

The proposed elevation for the RB foundation is at elevation of approximately 53 m CGD, corresponding to a depth of about 35 m below grade. At this elevation/depth, the Lindsay Formation has Rock Quality Designation values ranging from 90% to 100% and discontinuity spacing is considered to be 1 m to 3 m, per the 2022 Power Block geotechnical investigations (Reference 2.7-39).

Considering a mean UCS of 75 MPa and 48 MPa (Reference 2.7-39), the allowable bearing capacity ( $q_a$ ) for the RB is 7.5 MPa and 4.8 MPa, respectively.

For a conservative bearing capacity estimate, using a minimum UCS of 48 MPa and bearing capacity factor ( $K_{sp}$ ) of 0.1, an allowable bearing capacity of 4.8 MPa will be used for the Reactor Building foundation design.

#### ***2.7.5.1.2.3 Pile Foundation***

Pile foundations may also be considered for other heavily loaded power block structures. These structures may be supported on drilled caissons founded on competent undisturbed very dense/hard glacial till (with minimum 1 m embedment) or bedrock (with 1 m embedment recommended) with the over-excavation and backfill for soil deposits above elevation 80 m to 82 m CGD. End-Bearing Caissons founded on native undisturbed lower till deposit (Unit 5) at about 20 m depth can be designed for a factored geotechnical compression resistance 1100 kN. Alternatively, end-bearing caissons advanced to about 25 m depth, at least 1.0 m socket into bedrock (Unit 6a – Blue Mountain Formation), can be designed using a factored geotechnical compression resistance of 620 kN. The ultimate end-bearing resistance in bedrock is estimated to be approximately 20 MPa and a resistance factor of 0.4 is used to calculate the factored geotechnical compression resistance. These will form predominantly end-bearing foundations and therefore larger diameters (minimum 0.76 m in diameter) are recommended. Relatively undisturbed (clean) caisson bases should be ensured prior to concrete placement to minimize any potential settlement under maximum applied loads. The end-bearing caissons with at least 1 m embedment below weathered and/or fractured bedrock is estimated and presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

Uplift forces of cast-in place concrete caissons will be resisted by the weight of the foundation and friction along its embedment surface area. Estimation of uplift resistance of 1.0 m diameter caissons are presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

#### ***2.7.5.1.3 Earth Pressure***

The anticipated earth pressure considering the in-situ stress, ground conditions, soil shoring system, RB stiffness, and loads from surrounding buildings along the depth of the RB has been conservatively evaluated based on results of non-linear FIA, as presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) and is displayed in Figure 2.7.5.1.3-1.

The horizontal pressure was found higher in bedrock compared to the soil. This is due to the higher in-situ stress locked in the bedrock as a result of past tectonic activities. The earth pressure at the interface of the RB wall in the bedrock presented in Figure 2.7.5.1.3-1 represents a bounding post-construction stage scenario that assumes no stress release occurs in the bedrock

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

during excavation so all the in-situ stresses locked in the rock would be fully transferred to the RB wall. Reinforcement in the bedrock is to be incorporated in updates to the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) that estimates stress release in the bedrock at the end of the excavation once the rock reinforcement is designed. A field instrumentation plan is to be implemented, per guidance in Section 3.4 of NEDO-33914-A (Reference 2.7-27), to monitor the deformations of the rock during the excavation. These measurements will be used to calibrate the FIA model.

#### **2.7.5.1.4 Time-Dependent Deformation for Proposed Foundations**

##### **2.7.5.1.4.1 Elastic Settlement Method**

The elastic settlement is presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) and summarized in Table 2.7-5

**Table 2.7-5 Deformation for Proposed Foundations (Reference 2.7-38)**

<b>Building Structures</b>	<b>Structural Bearing Pressure, Upper Bound (kPa)</b>	<b>Proposed Foundation (Width, Depth) (m)</b>	<b>Estimated Elastic Settlement (mm)</b>
Control Building	28.7	Spread footing (3, 1.3)	1
Turbine Building	270	Raft Foundation (68X70, 1.3)	41
	150		23
	80		12
RAD Waste Building	162	Spread Footing (3, 1.3)	5
	162	Raft Foundation (48X30, 1.3)	16
Reactor Auxiliary Bay	36.8	Spread Footing (3, 1.3)	1

The expected settlement of raft foundation was analysed for the non-uniform structural load as documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

##### **2.7.5.1.4.2 Consolidation Settlement Method**

As detailed in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), it is anticipated that much of the consolidation settlement occurs in the lean clay deposit (Unit 4b). Given the Over-Consolidation-Ratio for Unit 4b is between 1.8 and 2.2, the lean clay deposit is over consolidated. Since the final effective pressure caused by the structural pressure is estimated to be lower than the pre-consolidation pressure in the deposit, the consolidation settlement is therefore estimated using the reconsolidation index (Cr). Annual secondary (creep) consolidation settlement is negligible.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

The estimated consolidation settlement of different building structures is summarized in Table 2.7-6.

**Table 2.7-6 Consolidation Settlement Method (Reference 2.7-38)**

<b>Building Structures</b>	<b>Structural Bearing Pressure, Upper Bound (kPa)</b>	<b>Proposed Foundation (Width, Depth) (m)</b>	<b>Estimated Consolidated Settlement (mm)</b>
Control Building	28.7	Spread Footing (3, 1.3)	5
Turbine Building	270	Raft Foundation (68X70, 1.3)	51
	150		31
	80		17
RAD Waste Building	162	Spread Footing (3, 1.3)	9
	162	Raft Foundation (48X30, 1.3)	45
Reactor Auxiliary Bay	162	Spread Footing (3, 1.3)	2

The expected settlement of raft foundation was analysed for the non-uniform structural load. The maximum total settlement (elastic and consolidated settlement) of the TB is approximately 92 mm, and the differential settlement is approximately 61 mm.

The settlement of raft foundations is also dependent on the rigidity of the foundation, homogeneity of the subgrade material and the construction method. Following the guidance of Section 4.0 of NEDO-33914-A (Reference 2.7-27), a 3-D non-linear FIA is to be performed to develop settlement contours of the raft foundations at a later design stage.

#### **2.7.5.2 Geotechnical and Seismological Site Design Parameters**

Subsection 2.7.5.2 presents the geotechnical and seismological properties for the seismic and structural analysis, and design, including:

- Subgrade profiles – Subsection
- Equivalent linearized static properties of soil and engineered fill materials – Subsection 2.7.5.2.2
- Equivalent linearized static properties of rock – Subsection 2.7.5.2.3
- Dynamic subgrade properties – Subsection 2.7.5.2.4
- Seismic Design Parameters – Subsection 2.7.5.2.5
- Groundwater Level – Subsection 2.7.5.2.6

#### **2.7.5.2.1 Subgrade Profiles Stratigraphy**

The design analyses of the deeply embedded BWRX-300 RB consider subgrade profiles to account for the variations of the soil and rock properties with depth at the DNNP site. The soil profiles represent “as-built” conditions at the DNNP site after construction of the BWRX-300 facility, where the engineered fill replaces the excavated top in-situ upper lacustrine or fill units. The stratigraphy of the as-built subgrade profiles consists of:

- Engineered fill that is for the upper 6 m to 8 m from elevation 80 m to 82 m CGD, as required to the final grade at elevation 88 m CGD.
- In-situ soils consisting of upper till (Unit 3), intermediate glaciolacustrine soils (Units 4a and 4b), and the lower till unit (Unit 5).
- Rock units including Blue Mountain (Unit 6a), Lindsay (Unit 6b), Verulam (Unit 6c), Bobcaygeon, Gull River, Shadow Lake and Genesis formations.

The engineered fill will comprise either commercial crusher run, or pit run granular fill or select excavated material meeting the requirements of engineered fill described under “Planned As-Built Soil Profile” in Subsection 2.7.3.2. Placement of the fill will be controlled based on in-situ testing and monitoring by the geotechnical engineer as described in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

The BWRX-300 RB vertical cylindrical shaft deep excavation is to be extended through the Blue Mountain Formation (Unit 6a) and founded in the Lindsay Formation (Unit 6b). The Gneiss formation – the deepest investigated unit - is taken as the hard rock basement with shear wave velocities that are greater than or equal to 3000 m/s, per the 2012 Field Work – Geology and Geological Evaluation NK054-REF-01210-0418696 (Reference 2.7-28).

The pre-excavation in-situ site stratigraphy for soil layers are presented in Table 2.7-2. The adopted in-situ soil layer thicknesses are based on the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39).

The stratigraphy of the rock units at the DNNP site including rock formations and thicknesses are presented in Table 2.7-3. The bedrock stratigraphy is based on the discussion presented in Subsection 2.7.3.2. The elevation of top of upper rock unit, the Blue Mountain (Whitby) Formation, considered as “top of rock” is expected to be about 64.2 m CGD with a variability of  $\pm 2$  m. The variation in the thickness layer of  $\pm 3$  m is based on the results of the 2022 DNNP Geotechnical Investigation reported in NK054-REP-01210-00175 (Reference 2.7-39).

#### **2.7.5.2.2 Equivalent Linearized Static Properties of Soil and Engineered Fill Materials**

Upper Bound and Lower Bound equivalent linearized properties representing the pressure of the soil and rock materials under long-term (static) loads are established based on measurements obtained from the different field and laboratory tests executed during the 2022 geotechnical investigation NK054-REP-012010-00175 (Reference 2.7-39). Upper and lower values are directly from the measured values. Further statistical analysis is completed to account for uncertainty as required during detailed design.

The static Elastic Modulus  $E_{st}$  values for soil materials are obtained from the results of field and laboratory tests. Initial Tangent Elastic Modulus values for the soil materials are established by Triaxial Compression Testing and Pressuremeter Testing, respectively. Initial Tangent Elastic Modulus is interpreted from consolidated anisotropic drained triaxial testing of reconstituted specimen. This is representative of in-situ conditions where the specimen is consolidated to approximate in-situ vertical effective stress.

Poisson's ratio  $\nu_{st}$  values are determined by compression and shear wave velocities measured during triaxial compression testing. Effective friction angle and Coefficient Lateral Earth Pressure at Rest is determined by Triaxial Compression Testing and Pressuremeter Testing during the 2022 geotechnical investigation NK054-REP-01210-000175 (Reference-2.7-39).

A summary of linearized static properties for engineered fill and in-situ soil layers in the as-built profiles are provided in the Table 2.7-7.

#### ***2.7.5.2.3 Equivalent Linearized Static Properties of Rock***

The 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.7-39) studied the linearized static properties of rock on the DNNP site, with focus around the BWRX-300 power block area. The linearized  $E_{st}$  and  $\nu_{st}$  values of the rock masses are evaluated from UCS testing and triaxial compression testing. The intact rock modulus was measured through UCS testing of intact rock samples. The intact rock modulus was then adjusted to evaluate the rock mass deformation modulus by two different methods:

- Evaluation of the Geologic Strength Index (two separate ways) and further calculation
- Directly measured through pressuremeter testing.

Total or bulk unit weight was measured during the uniaxial and triaxial compressive strength testing. Table 2.7-8 presents a summary of linearized static properties for the rock layers.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.7-7: As-Built Linearized Static Properties for Soil Layers (Reference 2.7-39)**

Stratigraphic Unit	Layer Thickness (m)	Total Unit Weight (kN/m <sup>3</sup> )	Effective Friction Angle (degrees)		Elastic (Young's) Modulus (MPa)		Estimated Coefficient of Lateral Earth Pressure at rest	
		Ave.	Ave.	Range	Lower	Upper	Ave.	Range
Unit 2a	0.00 - 5.72	21.5	32 <sup>(a)</sup>	25 – 37 <sup>(a)</sup>	13	49	0.61 <sup>(a)</sup> - 0.67 <sup>(b)</sup>	0.41 – 0.95 <sup>(a)</sup> 0.65-0.68 <sup>(b)</sup>
Unit 2b		non-plastic	34 <sup>(b)</sup>	29 – 41 <sup>(b)</sup>	32	80	0.73 <sup>(b)</sup>	0.49 – 0.91 <sup>(b)</sup>
Unit 3	<1.00 - 13.1	24.3	37 <sup>(a)</sup> 41 <sup>(b)</sup>	36 – 38 <sup>(a)</sup> 31 - 48 <sup>(b)</sup>	31	613	0.69 <sup>(b)</sup>	0.53 – 1.02 <sup>(b)</sup>
Unit 4a	0.00 - 17.7	22.1	40 <sup>(a)</sup> 39 <sup>(b)</sup>	39 – 41 <sup>(a)</sup> 35 - 45 <sup>(b)</sup>	52	600	0.57 <sup>(b)</sup>	0.42 – 0.73 <sup>(b)</sup>
Unit 4b		22.2	30 <sup>(a)</sup> 34 <sup>(b)</sup>	30 <sup>(a)</sup> 29 – 42 <sup>(b)</sup>	136	413	0.83 <sup>(a)</sup> 0.53 <sup>(b)</sup>	0.43 – 1.15 <sup>(a)</sup> 0.34 - 0.70 <sup>(b)</sup>
Unit 5	0.00 – 6.4	23.7	35 <sup>(a)</sup> 31 <sup>(b)</sup>	32-38 <sup>(a)</sup> 26 – 36 <sup>(b)</sup>	110	330	0.58 <sup>(a)</sup> 0.49 <sup>(b)</sup>	0.39 – 0.74 <sup>(a)</sup> 0.39-0.58 <sup>(b)</sup>

(a) From Triaxial Compression Testing

(b) From Pressuremeter Testing



**Table 2.7-8: Summary of Linearized Static Rock Properties (Reference 2.7-39)**

Stratigraphic Unit	Average Bulk Density (kgN/m <sup>3</sup> )	Mean Intact Modulus (GPa)	Rock Mass Deformation Modulus (GPa)		Poisson's Ratio
			GSI	Pressuremeter Tests	
Unit 6a - Blue Mountain Shale / Shale+Limestone / Limestone	2641	26.6	17.9	5.91	0.32/0.28/0.00
Unit 6b – Lindsay Formation Shale / Shale+Limestone / Limestone	2681	43.4	38.7	9.75	0.00/0.22/0.36
Unit 6c – Verulam Formation Shale / Shale+Limestone / Limestone	2679	25.0	22.3	12.29	0.21/0.29/0.25

#### ***2.7.5.2.4 Dynamic Subgrade Properties***

The measured values for dynamic properties of rock are presented in Table 2.7-9a and Table 2.7-9b. The measured small-strain in-situ soil dynamic properties are listed in Table 2.7-10a and Table 2.7-10b. The compression wave velocities, shear wave velocities for the soil and bedrock rock units are obtained from the measurements during the 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.7-39). Poisson's Ratio, Young's Modulus, Shear Modulus and Bulk Modulus are presented as calculated in the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39).

**Table 2.7-9a: Rock Dynamic Properties (Reference 2.7-39)**

Stratigraphic Unit	Total Unit Weight (kN/m <sup>3</sup> )	Shear Wave Velocity (m/s)		Compression Wave Velocity (m/s)		Poisson's Ratio	
		Average	Range	Average	Range	Average	Range
Unit 6a – Blue Mountain Formation	26.4	2405	1841 - 2953	4283	3073 - 5935	0.26	0.20 - 0.38
Unit 6b -Lindsay Formation	26.6	2640	1934 - 3024	4792	3202 - 5773	0.28	0.27 - 0.30
Unit 6c – Verulam Formation	26.6	2559	2128 - 2801	4570	3772 - 5443	0.27	0.26 - 0.28

**Table 2.7-9b: Rock Dynamic Properties (Reference 2.7-39)**

Stratigraphic Unit	Dynamic Shear Modulus (MPa)		Dynamic Young's Modulus (MPa)		Dynamic Bulk Modulus (MPa)	
	Average	Range	Average	Range	Average	Range
Unit 6a – Blue Mountain (Whitby) Formation	15320	12772 - 18186	38674	34068 - 45959	0.26	0.20 - 0.38
Unit 6b – Lindsay Formation	18696	17099 - 19458	47931	43539 - 49978	0.28	0.27 - 0.30
Unit 6c – Verulam Formation	17544	16534 - 18041	44614	42363 - 45762	0.27	0.26 - 0.28

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.7-10a: In-situ Soil Dynamic Properties (Reference 2.7-39)**

Stratigraphic Unit	Total Unit Weight (kN/m <sup>3</sup> )	Shear Wave Velocity (m/s)		Compression Wave Velocity (m/s)		Poisson's Ratio	
		Average	Range	Average	Range	Average	Range
Unit 2a – Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	21.5	304	215 - 451	1087	560 - 2200	0.43	0.15 - 0.48
Unit 2b – Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt	0.00	351	255 - 483	1769	800 - 2200	0.48	0.45 - 0.49
Unit 3 – Upper Till	24.3	489	240 - 705	1845	700 - 2400	0.46	0.42 - 0.48
Unit 4a – Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt	22.1	659	362 - 1078	2107	1600 - 2400	0.44	0.30 - 0.48
Unit 4b – Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	22.2	656	440-994	2118	1800-2400	0.44	0.37-0.47
Unit 5 (Lower Till)	23.7	875	683-1344	2470	2000-3400	0.42	0.24-0.47

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.7-10b: In-situ Soil Dynamic Properties (Reference 2.7-39)**

Stratigraphic Unit	Dynamic Shear Modulus (MPa)		Dynamic Young's Modulus (MPa)		Dynamic Bulk Modulus (MPa)	
	Average	Range	Average	Range	Average	Range
Unit 2a – Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	198	94-415	550	277-950	2374	446 - 6271
Unit 2b – Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt	298	181-450	879	525-1331	7304	1658 - 10750
Unit 3 – Upper Till	577	265 - 878	1678	783 - 2489	7340	4066 - 10472
Unit 4a – Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt	1025	454 - 2607	2915	1338 - 6752	8636	5497 - 10914
Unit 4b – Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	947	432 - 1808	2719	1272 - 4941	8386	6038 - 10869
Unit 5 (Lower Till)	1848	1133 - 3670	5217	3273 - 10435	12180	5093 - 22215

#### **2.7.5.2.5 Seismic Design Parameters**

Two sets of seismic design parameters were developed based on the results of the site response analyses described in Subsection 2.7.4.6.1 for DBE seismic design and BDBE DEC seismic evaluations:

- Ground motion spectra defining the amplitude and frequency content of the DBE and BDBE ground motion at the DNNP site
- Hazard-Consistent, Strain-Compatible (HCSC) profiles defining the variation with depth of the dynamic properties of DNNP subgrade materials compatible to the strains generated by DBE and BDBE levels

##### **2.7.5.2.5.1 Ground Motion Spectra**

Per guidance of NEDO 33914-A, Section 5.2.2, three sets of response spectra are developed, as described in Section 9.2 of the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41), defining the amplitude and frequency content of the DNNP site-specific DBE and BDBE ground motion:

1. Foundation Input Response Spectra at RB foundation bottom elevation of 52.93 m CGD presented in Figure 2.7.5.2.5-1
2. Performance Based Intermediate Response Spectra at the soil / rock interface elevation 64 m CGD, located 24 m below planned finished grade presented in Figure 2.7.5.2.5-2
3. Performance Based Surface Response Spectra at the finished grade elevation of 88 m CGD presented in Figure 2.7.5.2.5-3

##### **2.7.5.2.5.2 Strain-Compatible Soil Properties**

Profiles of HCSC dynamic subgrade properties, needed for the SSI analyses, are developed based on the results from the site response analyses described in Section 2.7.4.6. The profiles defining the variation with depth of subgrade shear wave velocities compatible to the DBE and BDBE strain levels are presented in Figure 2.7.5.2.5-4. The profiles of subgrade compression wave velocities for the DBE and BDBE strain levels are presented in Figure 2.7.5.2.5-5. Figure 2.7.5.2.5-6 presents the subgrade damping profiles representing the dissipation of energy in the subgrade materials for the DBE and BDBE levels. The presented HCSC dynamic subgrade properties are per Section 9.3 (Table 9-40 through Table 9-45) of the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41).

##### **2.7.5.2.6 Groundwater Level**

The groundwater elevations are listed in Table 3-7 in Volume 2 of 2 of the 2022 Phase 1 Geotechnical Investigation - DNNP (Reference 2.7-39), and is replicated, in part, in Table 2.7-11, which provides samples of groundwater elevation and hydraulic vertical gradient at BH12 area which is located on the western side of the RB perimeter, as shown in Figure 2.7.3.1-6.

**Table 2.7-11 Samples of Groundwater Elevation and vertical Hydraulic Gradients, BH12 Area (Reference 2.7-39, Table 3-7)**

Date	Groundwater Elevation (m)				Vertical Gradient (m/m)		
	Unit 3 <sup>1</sup>	Unit 4a <sup>2</sup>	Unit 5 <sup>3</sup>	Unit 6 <sup>4</sup>	Unit 3 to 4a (down)	Unit 4a to 5 (Down)	Unit 5 to 6 (Down)
	BH12-1	BH12-2	BH12-3	BH14			
<b>29NOV21</b>	85.47	83.89	79.47	77.09	-0.25	-0.71	-0.28
<b>05JAN22</b>	85.72	84.03	82.47	78.56	-0.27	-0.25	-0.46
<b>07FEB22</b>	85.24	83.66	79.18	78.41	-0.25	-0.72	-0.09
<b>17FEB22</b>	85.46	83.81	79.29	78.52	-0.26	-0.73	-0.09

1. Upper Till
2. Surficial Glaciolacustrine Deposits
3. Lower Till
4. Bedrock

Based on the groundwater information at the DNNP site presented in Subsection 2.7.3.2 and Table 2.7-11, an upper bound groundwater level at elevation of 85.74 m CGD (or approximately 86 m CGD) corresponding to a depth of 2.26 m (or approximately 2 m) below the plant grade at elevation 88 m CGD is to be used for design.

### 2.7.5.3 Geotechnical Variability and Uncertainty

Geotechnical variability and uncertainty are considered in detail in the 2022 Geotechnical Investigation NK054-REP-01210-00175 (Reference 2.7-39).

When sampling the soil and rock there can be sampling bias that is introduced in the sample selection process. In general, DNNP project samples were selected based on predetermined testing requirements for each borehole and samples were selected from a variety of depths within each borehole. In some cases, such as the shale from the Blue Mountain Formation, it is not possible to test the weaker rock as intact samples of this material cannot properly be prepared for testing (typically breaking apart along weaker bedding planes). In these cases, sensitivity analysis and engineering judgement are required during design to account for the fact that the range in the data may not capture the minimum values.

When in-situ and laboratory methods are used to measure soil and rock attributes, the inherent variability along with measurement error typically led to data scatter. Measurement error may result from equipment errors and procedural or operator errors. Measurement error is minimized through equipment calibration, standardized procedures, laboratory accreditation, etc.

In-situ and laboratory methods are also subject to statistical uncertainty, which may be reduced by increasing the sampling frequency. Further, certain in-situ and laboratory measurements are transformed for design purposes through empirical or other correlation methods.

Geotechnical variability and uncertainty are addressed using a two-pronged approach:

- Reduction in uncertainty through the use of reliable, calibrated equipment, precision in measurement and testing procedures and sufficient quantity of sampling/testing.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- Consideration of total variability associated with each geotechnical property/parameter, including evaluation of statistical parameters and identification of sources of uncertainty particular to each property/parameter.

**2.7.6 References**

- 2.7-1 NK054-REP-01210-00011 R001, 2009. "Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.7-2 Report No. 89575, 1989, "Darlington NGS - Preoperational Summary Report on Site
- 2.7-3 Seismic Hazard in Southern Ontario, Final Report, 1997 "Part 1: Seismic Source Models, Recurrence Models and Ground Motion Attenuation Models; and Part 2: Seismic Hazard Results and Sensitivity, prepared for the Atomic Energy Control Board of Canada,". Geomatrix Consultants, Inc.
- 2.7-4 NK054-REP-01210-00014 R001, 2009, "Site Evaluation of The OPG New Nuclear at Darlington – Probabilistic Seismic Hazard Assessment," Ontario Power Generation.
- 2.7-5 CNSC Regulatory Document RD-346, "Site Evaluation for New Nuclear Power Plants."
- 2.7-6 CNSC Regulatory Document REGDOC-1.1.1, "Site Evaluation and Site Preparation for New Reactor Facilities."
- 2.7-7 IAEA Safety Standard Series No. NS-R-3, "Site Evaluation for Nuclear Installations Safety Requirements." International Atomic Energy Agency.
- 2.7-8 IAEA Safety Standards Series No. SSR-1, "Site Evaluation for Nuclear Installations Safety Requirements," International Atomic Energy Agency.
- 2.7-9 NK38-REP-03611-10041 R000, 2011, "Update of The OPG Darlington Site Probabilistic Seismic Hazard Assessment for The Darlington Risk Assessment (DARA)," Ontario Power Generation.
- 2.7-10 NK38-REP-03611-10041 R002, 2019 Update of The OPG Darlington Site Probabilistic Seismic Hazard Assessment for The Darlington Risk Assessment (DARA)," Ontario Power Generation.
- 2.7-11 NK38-REP-03611-10041 R003, 2021, "Update of The OPG Darlington Site Probabilistic Seismic Hazard Assessment for The Darlington Risk Assessment (DARA)," Ontario Power Generation.
- 2.7-12 EPRI TR-102261-V1, 1994, "The Earthquakes of Stable Continental Regions, Volume 1; Assessment of Large Earthquake Potential," Electric Power Research Institute.
- 2.7-13 Canadian Journal of Earth Sciences, Volume 3, No. 5, " 1966, The St. Lawrence Valley System: A North American Equivalent of the East African Rift Valley System" Canadian Science Publishing.
- 2.7-14 Wheeler, R.L., "Sources of Seismic Hazard near the Eastern Great Lakes," Geological Society of America Abstracts with Programs, Northeastern Section, p. 110, 1996.
- 2.7-15 Slemmons, D.B., et al., "Neotectonics of North America: The Seismicity and Seismotectonics of Eastern Canada," Geological Society of America, 1991.
- 2.7-16 Thomas, R.L., et al., "Recent Deformation in the Bottom Sediments of Western and Southeastern Lake Ontario and Its Association with Major Structures and Seismicity" Geographie physique et Quaternaire, v. 47, no. 3:325-335. 1993.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- 2.7-17 Wheeler, R.L., "Earthquakes and the Cratonward Limit of Iapetan Faulting in Eastern North America" *Geology*, v. 23, no. 2:105-108, 1995.
- 2.7-18 Milkereit, B., et al., "Seismic Images of a Grenvillian Terrane Boundary" *Geology*, v. 20:1027-1030. 1992.
- 2.7-19 Adams, J., Allen J., Halchuk, S., and Kolai, M, "Canada's 6<sup>th</sup> Generation Seismic Hazard Model as Prepared for the 2020 National Building Code of Canada," 12<sup>th</sup> Canadian Conference on Earthquake Engineering, 2019.
- 2.7-20 US NRC NUREG/CR-6728, 2001, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines."
- 2.7-21 US NRC NUREG-2115, 2012, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," PEER NGA-East Model (Goulet et al).
- 2.7-22 Goulet, C., et. Al, "Central and Eastern North America Ground-Motion Characterization NGA-East Final Report," Pacific Earthquake Engineering Research Center, PEER Report No. 2018/08.
- 2.7-23 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of The OPG New Nuclear at Darlington – Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.7-24 NK38-02004P, 1981, "Darlington GS Geological Mapping of Bedrock Excavations," Ontario Power Generation.
- 2.7-25 Hydro Geotechnical Engineering Dept. Report 77110, "The Geology and Seismicity of the Darlington G.S. Site," 1977, Ontario Power Generation.
- 2.7-26 NK054-DRAW-01210-00003 R001, "Darlington Topographic Drawing," Ontario Power Generation.
- 2.7-27 NEDO-33914-A, Revision 2, 2022, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC.
- 2.7-28 NK054-REF-01210-0418696, 2012, "Geologic and Geophysical Evaluation, Darlington Site Investigation – Phase III (Field Work), AMEC Report No. D0053/RP/002 R01, Volumes 1 and 2," Ontario Power Generation.
- 2.7-29 NK054-REP-01210-00098 R000, 2013, Geotechnical Data Report – R2, Darlington New Nuclear Project Geotechnical Investigation, EXP Services Inc. Project No. BRM-00025482-A0," Ontario Power Generation.
- 2.7-30 NK054-REP-07730-00005 Rev. R000, 2009, Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.7-31 CSA Group, 2021, CSA N289.2:21, Ground motion determination for seismic qualification of nuclear power plants, January.
- 2.7-32 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.7-33 CSA Group, 2018, CSA N289.1-18, General requirements for seismic design and qualification of nuclear power plants, May.
- 2.7-34 CSA Group, 2020, CSA N289.3:20, Design procedures for seismic qualification of nuclear power plants, January.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- 2.7-35 U.S. Nuclear Regulatory Commission (USNRC), 2012a, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, Attachment 1 to Seismic Enclosure 1, March 12, ADAMS ML12053A340.
- 2.7-36 Electric Power Research Institute (EPRI), 2006, Program on Technology Innovation: Truncation of the Lognormal Distribution and Value of the Standard Deviation for Ground Motion Models in the Central and Eastern United States: TR – 1014381, August 2006, Palo Alto, California.
- 2.7-37 U.S. Nuclear Regulatory Commission (USNRC), 2010, DC/COL ISG-017, Interim Staff Guidance on Ensuring Hazard Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses (NRC ADAMS Accession Number ML092230543).
- 2.7-38 NK054-REP-03500.8-00003, 2023, “Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report,” Ontario Power Generation.
- 2.7-39 NK054-REP-01210-00175 R000, 2022, “Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project”, Volume 2 of 2 “Geotechnical Interpretation of Design Parameters,” Ontario Power Generation.
- 2.7-40 NK054-REP-10180-00001 R000, 2023, “Offshore Geotechnical Investigation Darlington New Nuclear Project,” Ontario Power Generation.
- 2.7-41 NK054-REP-03500.8-00001 R000, 2022, “Darlington New Nuclear Project Site-Specific Probabilistic Seismic Hazard Assessment”, Ontario Power Generation.
- 2.7-42 NK054-REP-03500.8-00002 R000, 2022, “Darlington New Nuclear Project - Seismically-Induced Soil Liquefaction Assessment,” Ontario Power Generation.
- 2.7-43 NK054-REP-01210-00142 R000, 2022, “Darlington New Nuclear Project – Site Evaluation Update Summary Report,” Ontario Power Generation.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

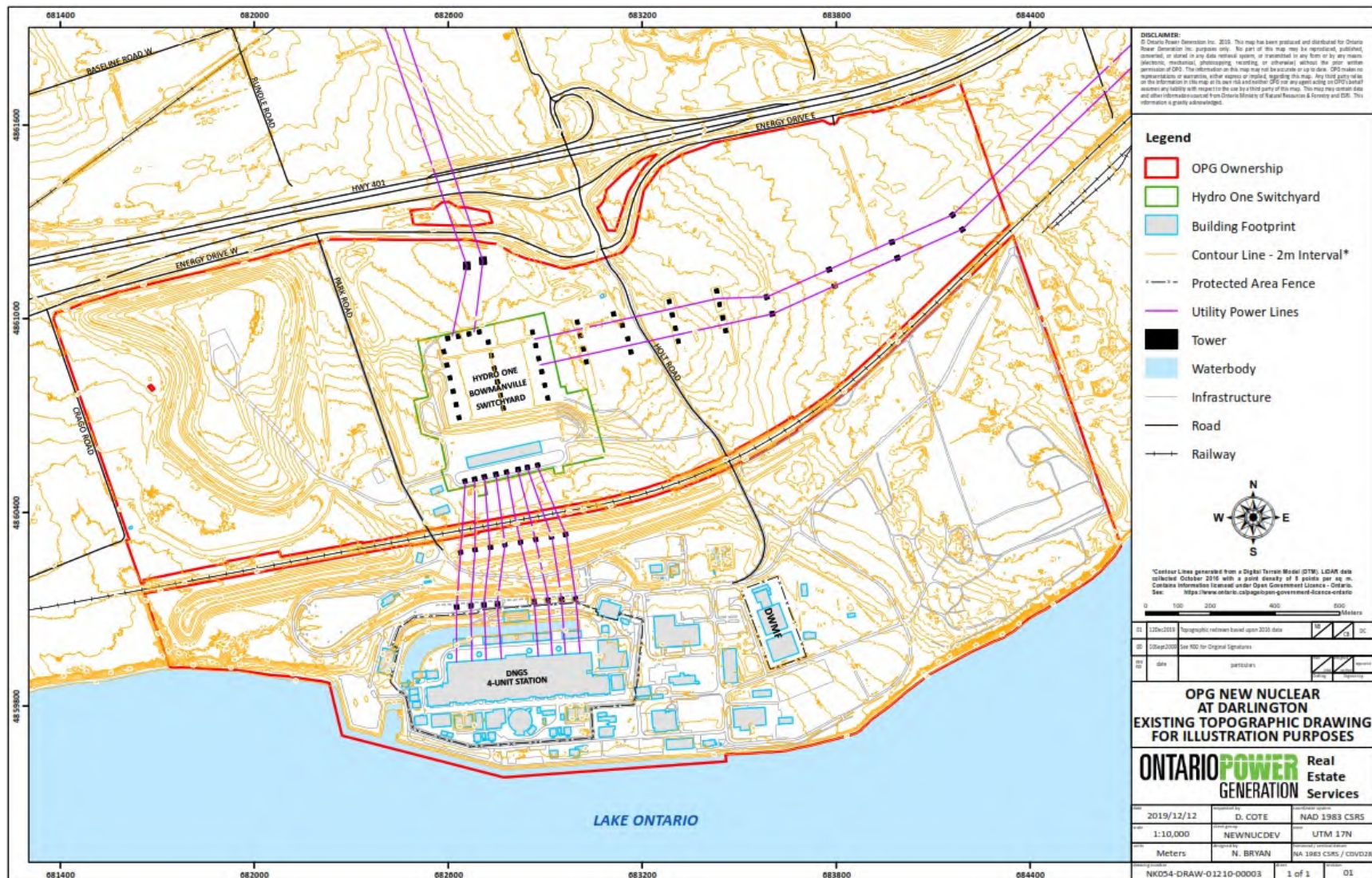


Figure 2.7.1-1: Existing Topographic Contours of DNNP Site (Reference 2.7-26)



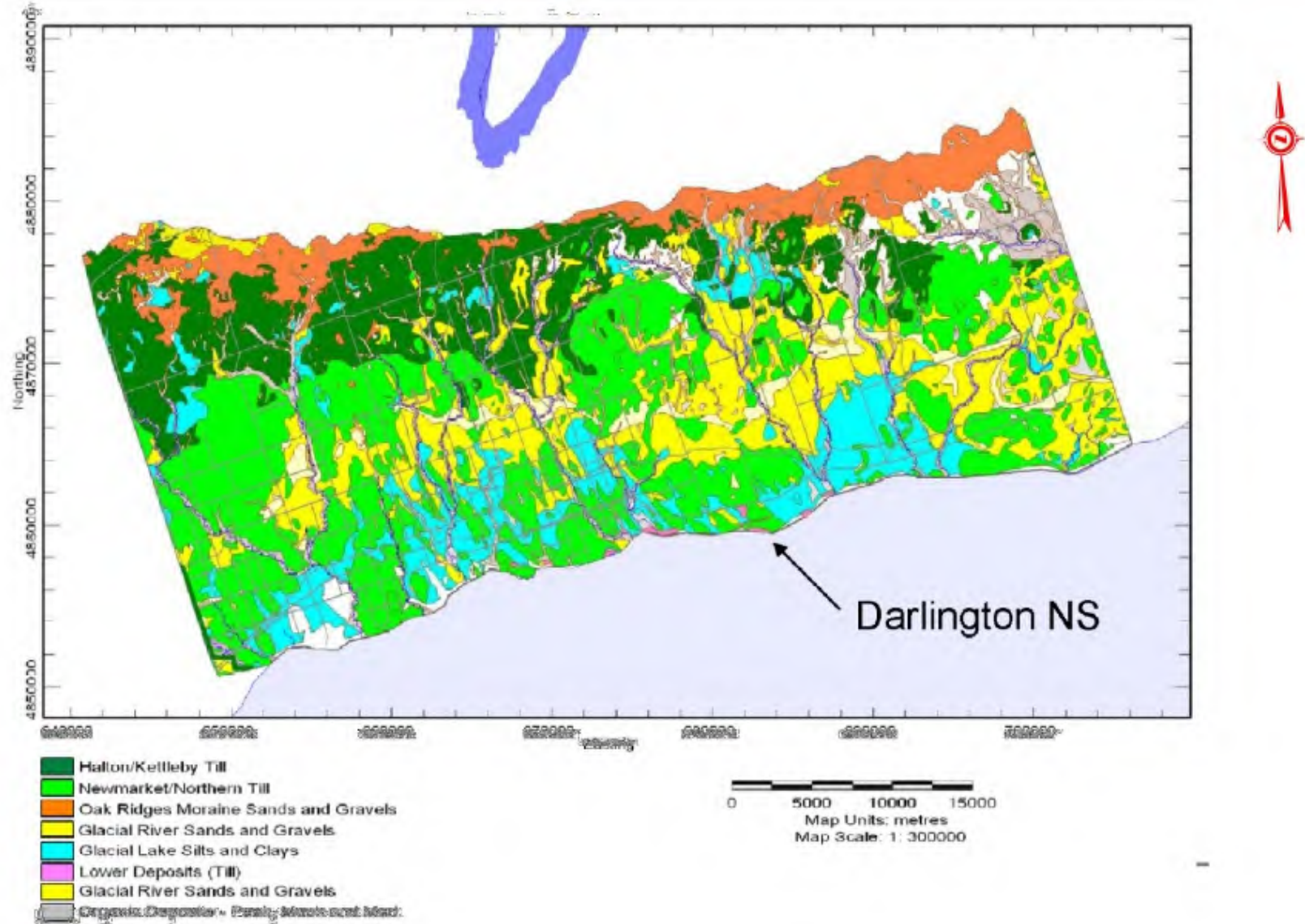


Figure 2.7.2-1: Darlington Nuclear Site - Regional Surficial Geology (Reference 2.7-1)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

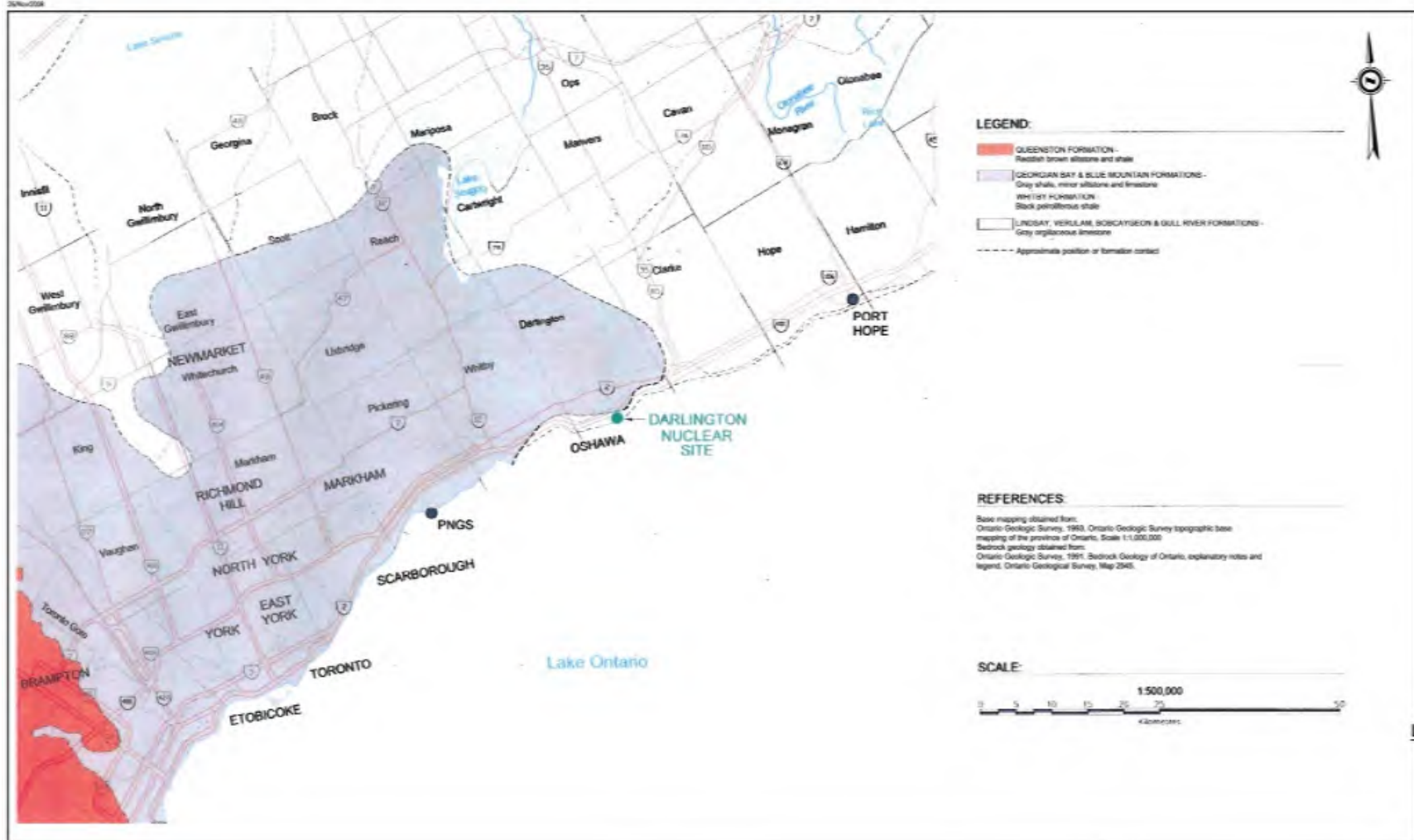


Figure 2.7.2-2: Darlington Nuclear Site - Regional Bedrock Geology (Reference 2.7-1)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

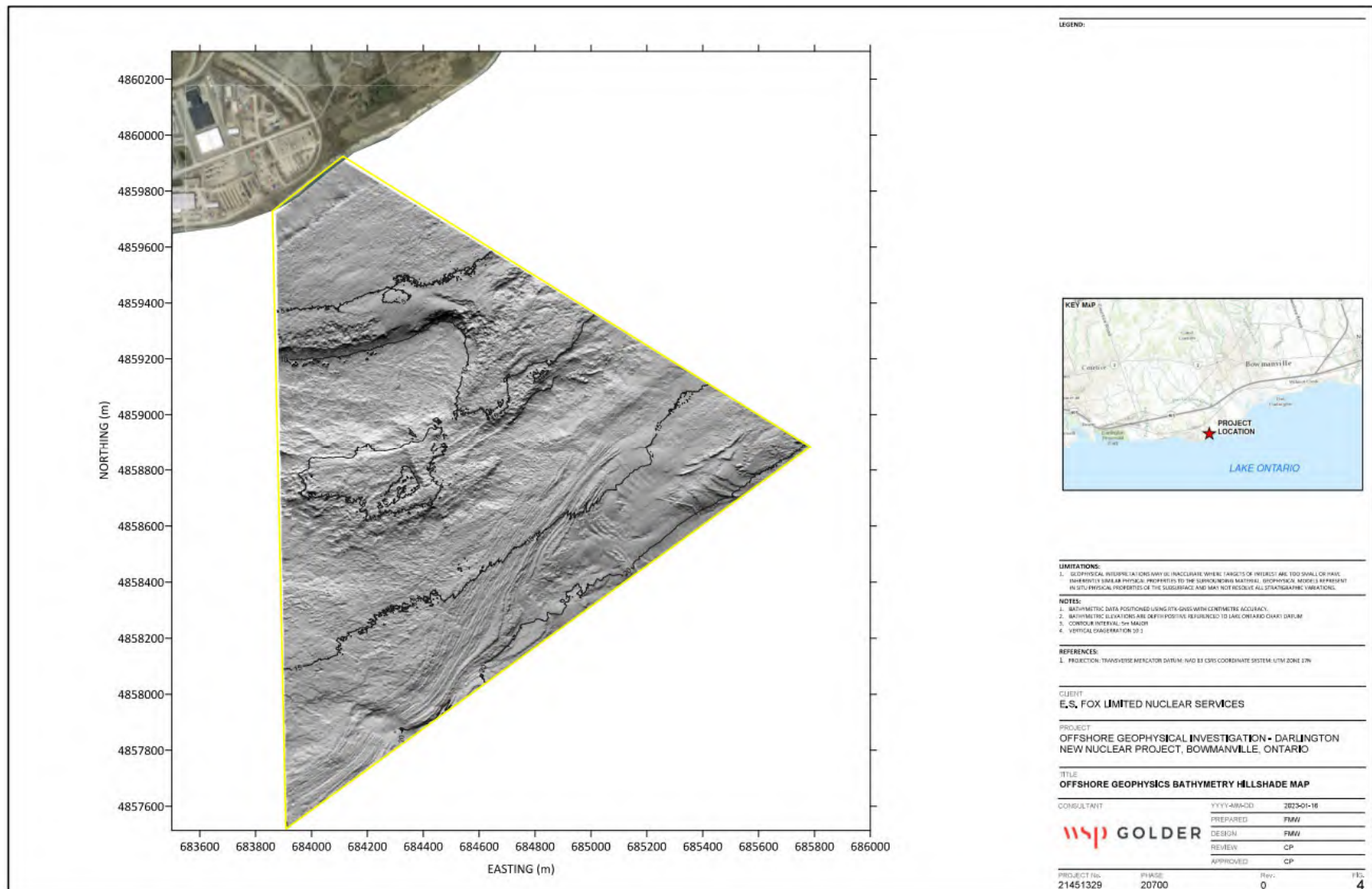


Figure 2.7.2-3: Offshore Geophysics Bathymetry Hillshade Map (Reference 2.7-40)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

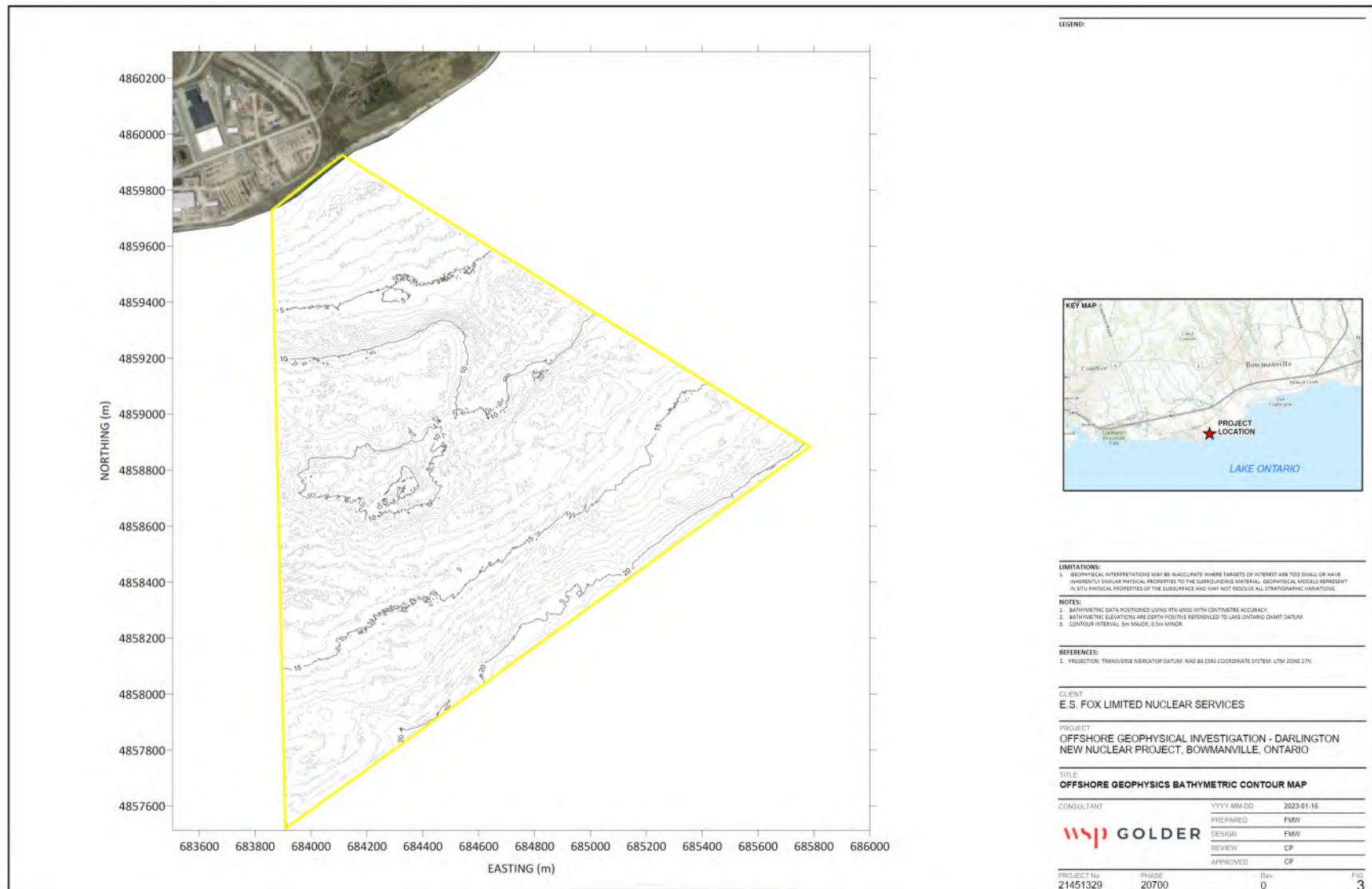


Figure 2.7.2-4: Offshore Geophysical Bathymetric Contour Map (Reference 2.7-40)

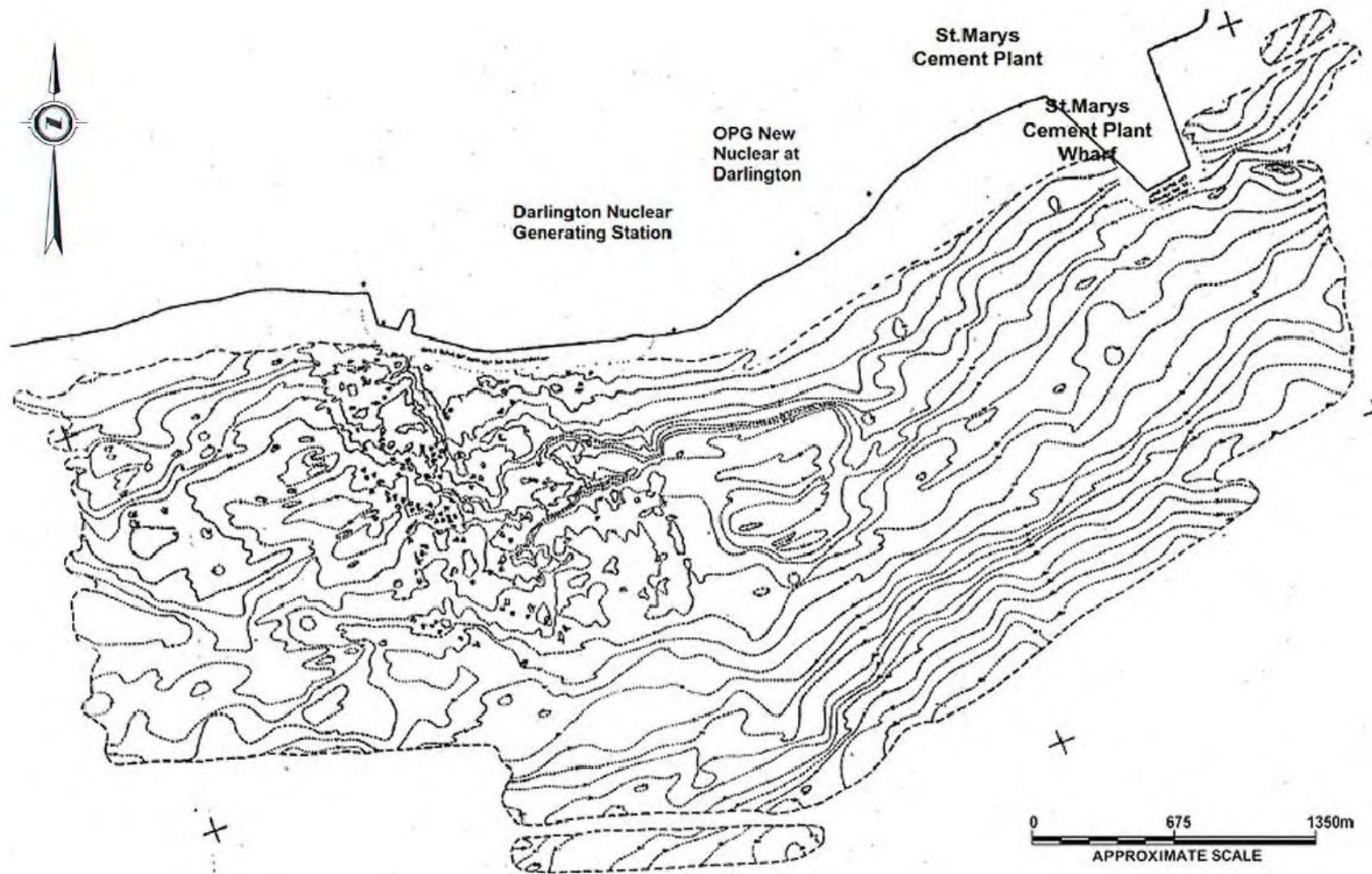


Figure 2.7.2-5: Lakebed Bathymetric Contours along DNNGS Site's Shoreline (Reference 2.7-2)



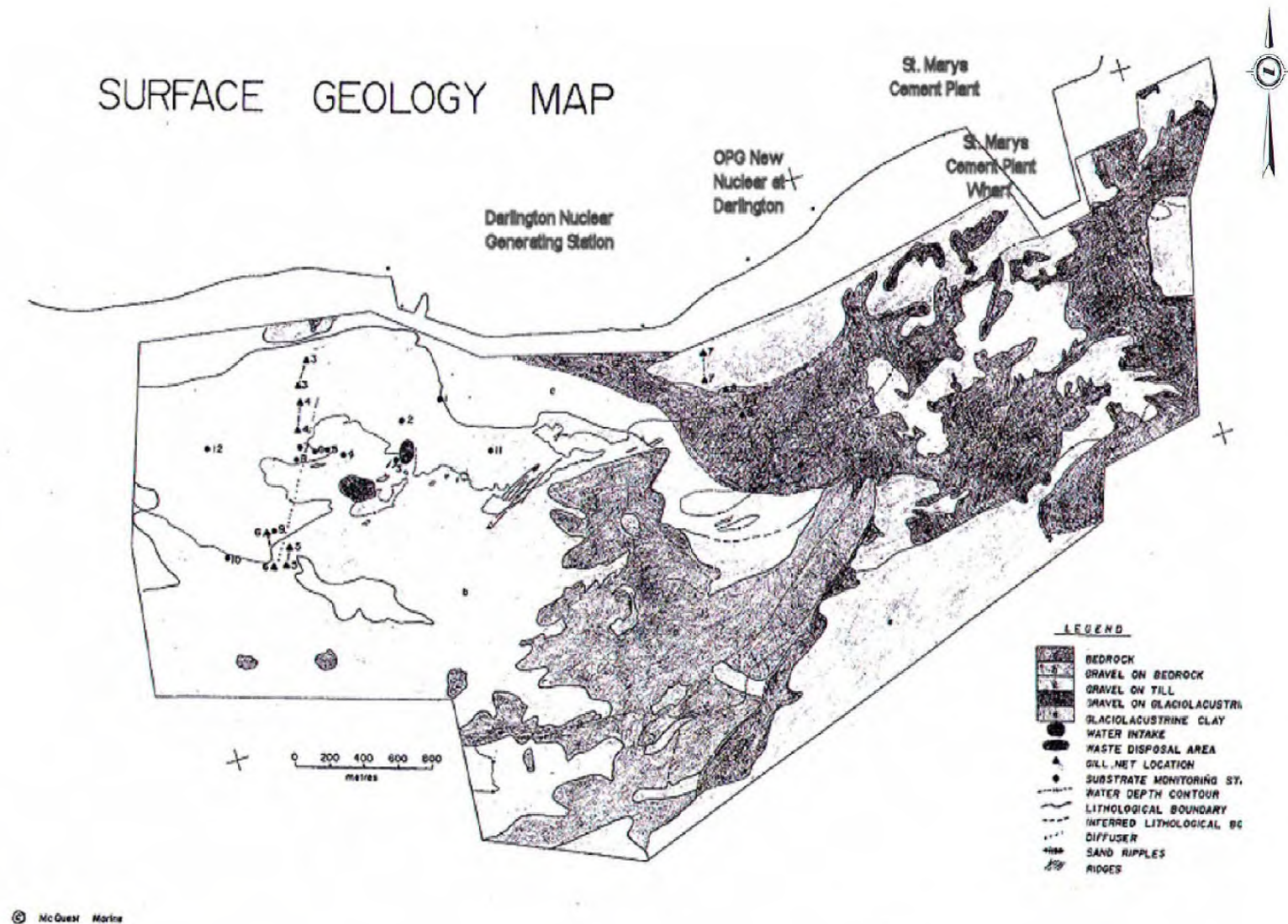


Figure 2.7.2-6: Lakebed Surface Geology Map along DNNP Site's Shoreline (Reference 2.7-2)



2-134

# NEDO-33951 REVISION 2 NON-PROPRIETARY INFORMATION



Figure 2.7.3.1-2: Locations of AMEC (2012) Boreholes (Reference 2.7-35)



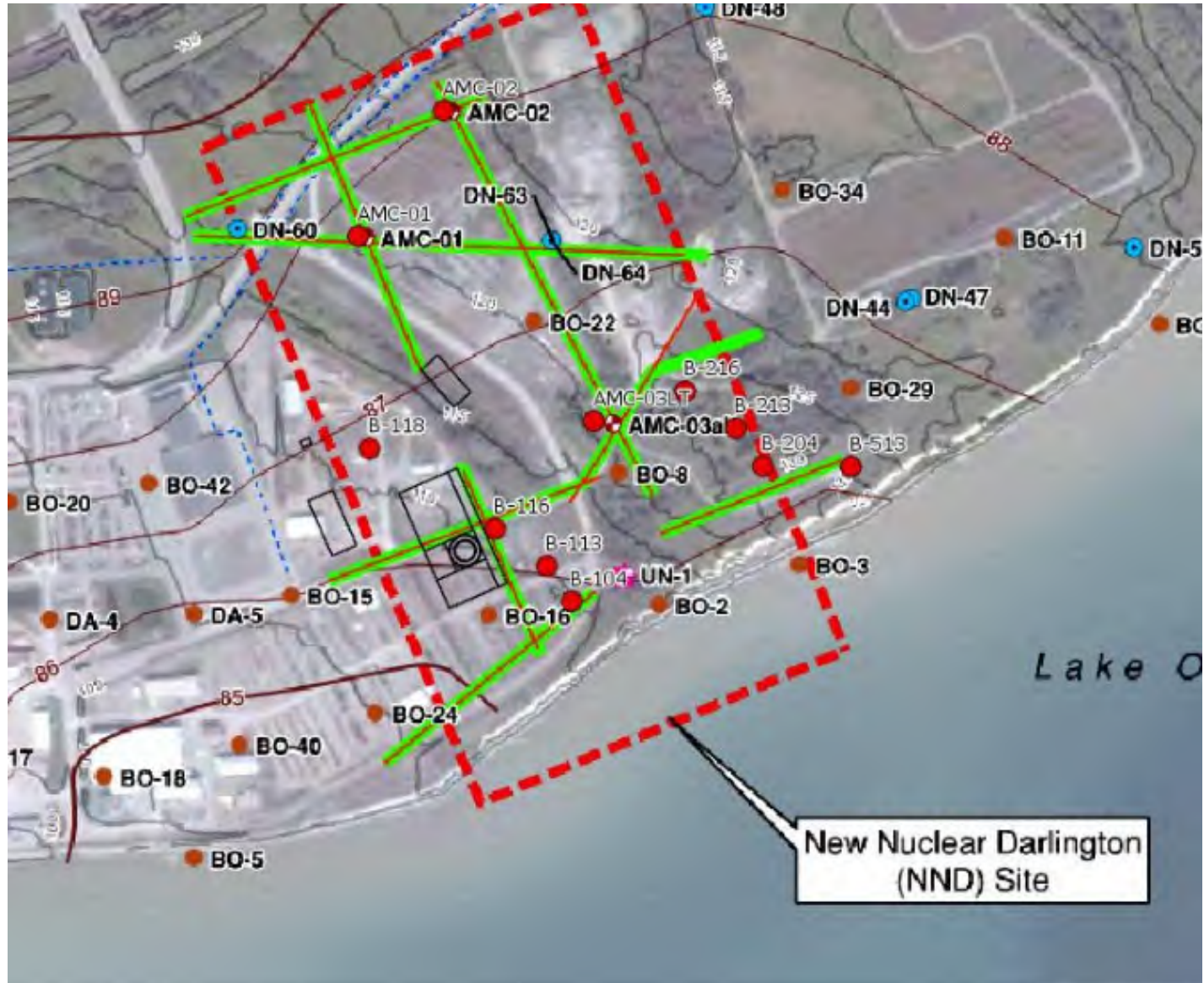
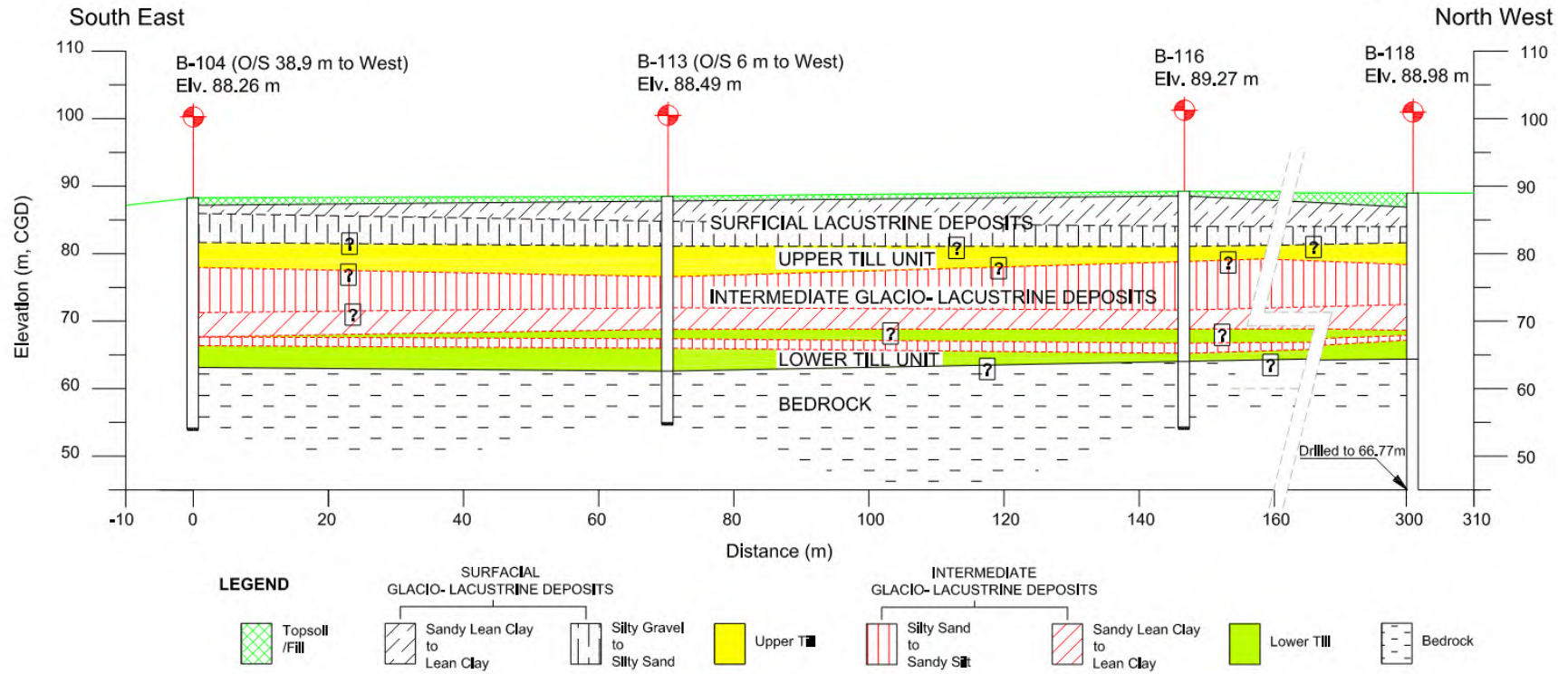


Figure 2.7.3.1-3: Locations of EXP (2013) Boreholes (Reference 2.7-36)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**METRIC**

DIMENSIONS ARE IN METRES  
AND/OR MILLIMETRES UNLESS  
OTHERWISE SHOWN. STATIONS  
ARE IN KILOMETRES + METRES.



**Figure 2.7.3.1-4: Stratigraphic Profile near BWRX-300 Location (Reference 2.7-36)**

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

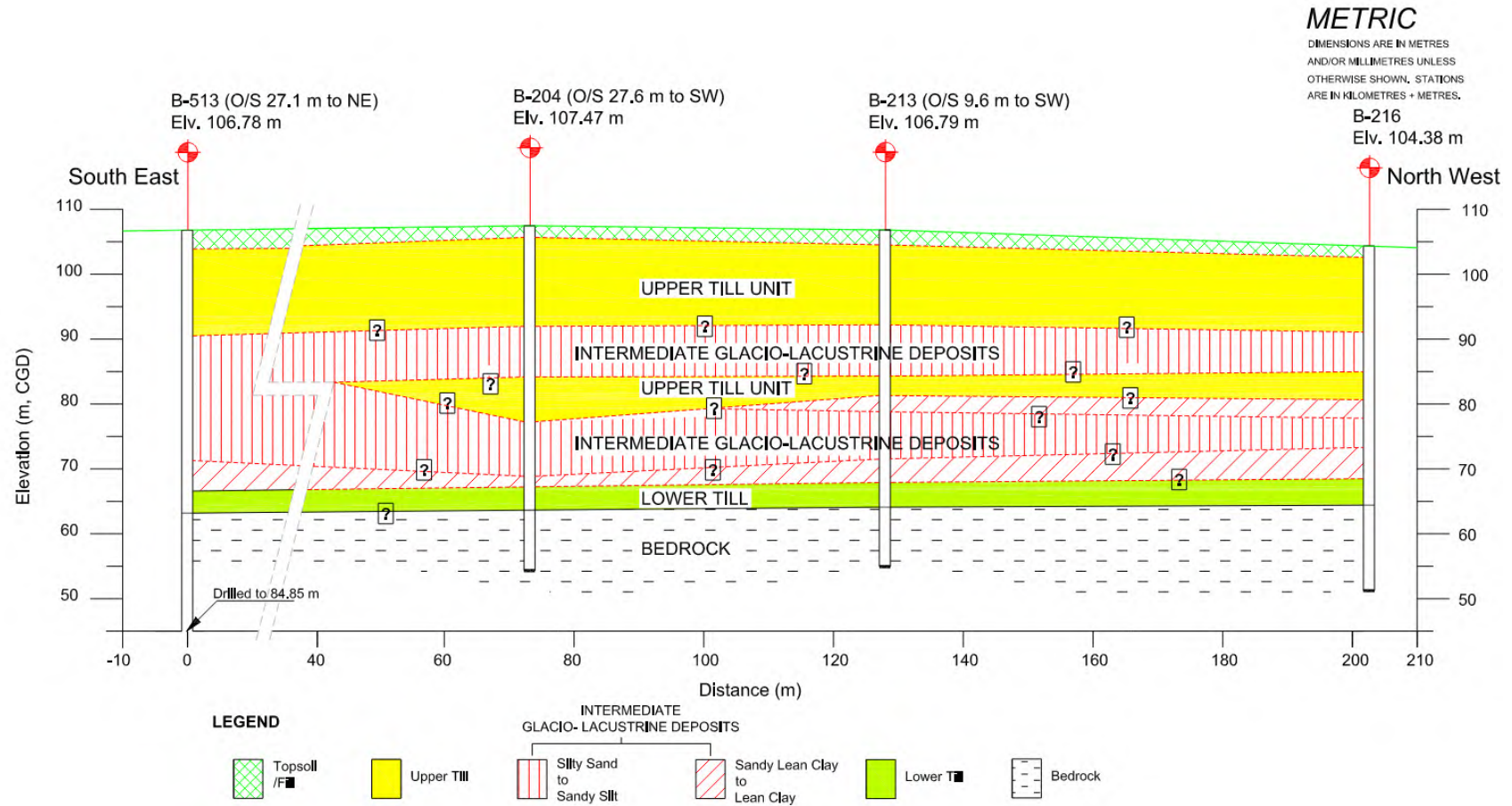


Figure 2.7.3.1-5: Stratigraphic Profile near Topographic Ridge East of the BWRX-300 Location (Reference 2.7-36)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

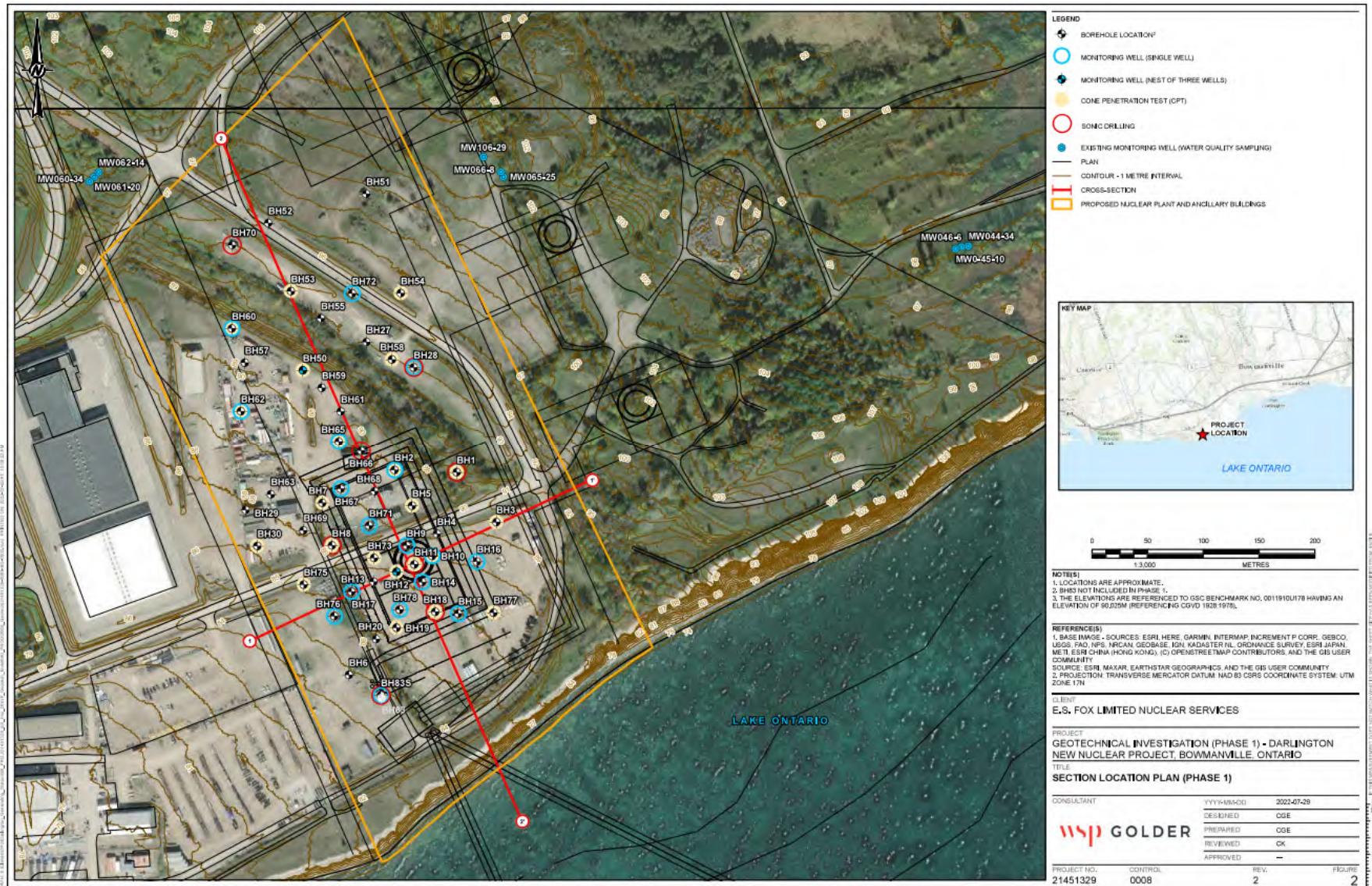


Figure 2.7.3.1-6: Boreholes Sections (1-1' and 2-2') Locations Plan (Phase 1) (Reference 2.7-39)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

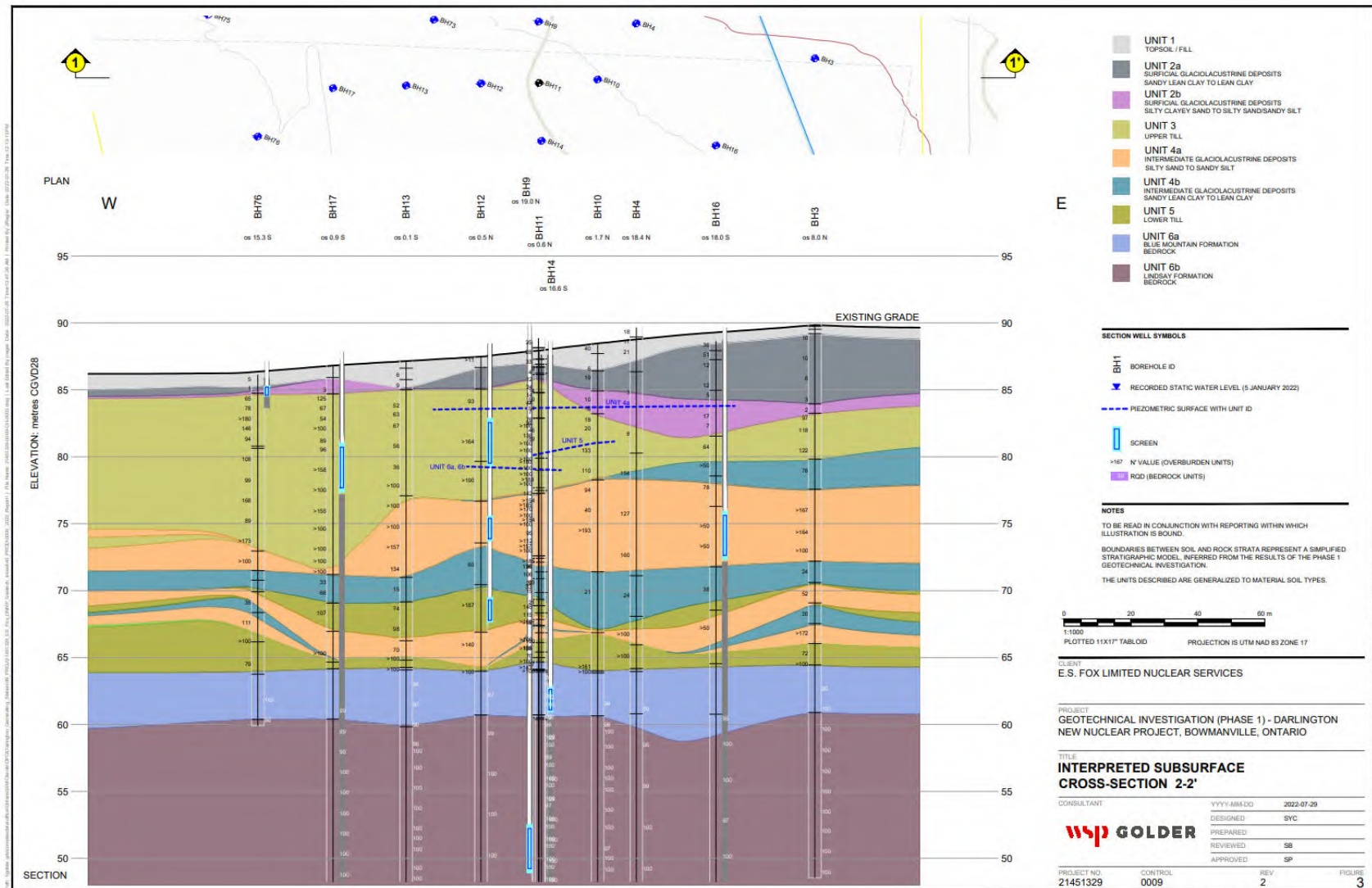


Figure 2.7.3.2-1: Subsurface Stratigraphic Profile at Cross-Section 1-1 (Reference 2.7-39)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

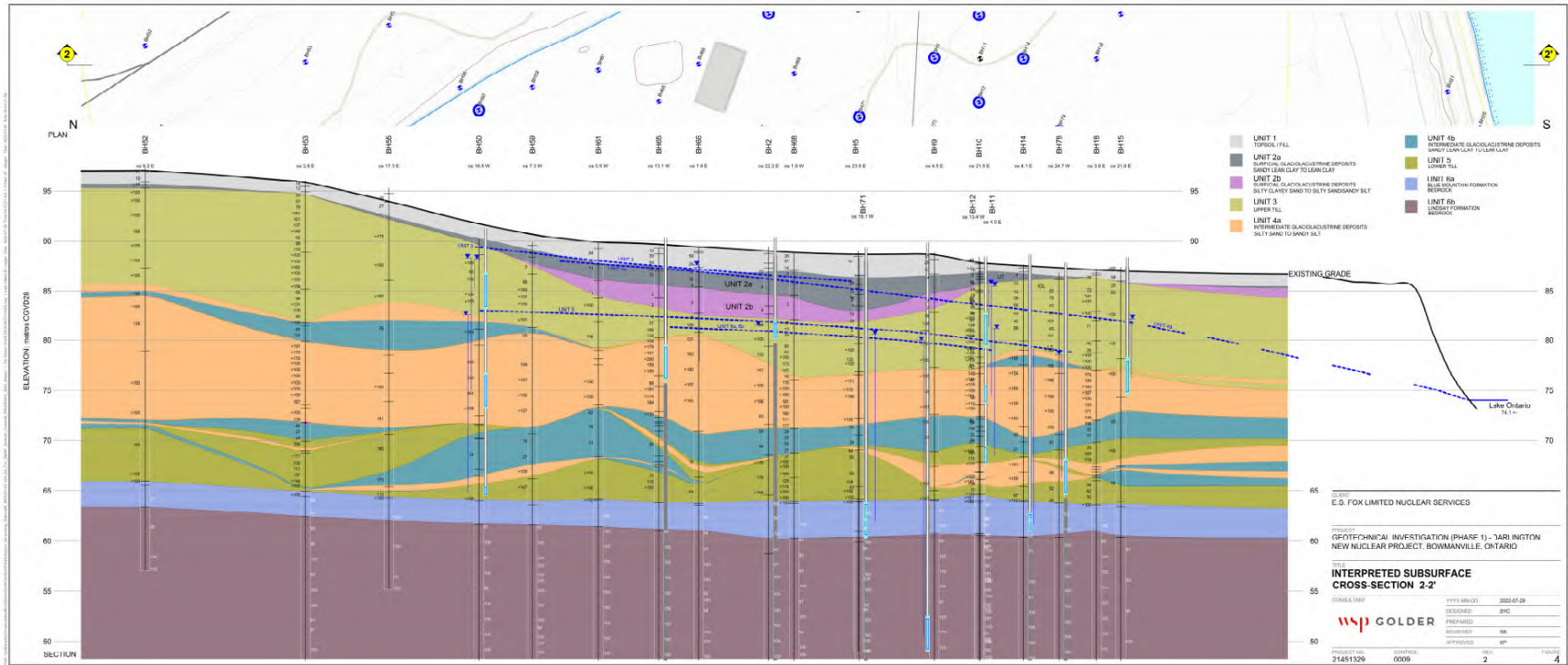


Figure 2.7.3.2-2: Subsurface Stratigraphic Profile at Cross-Section 2-2 (Reference 2.7-39)



[illegible]

2-142

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

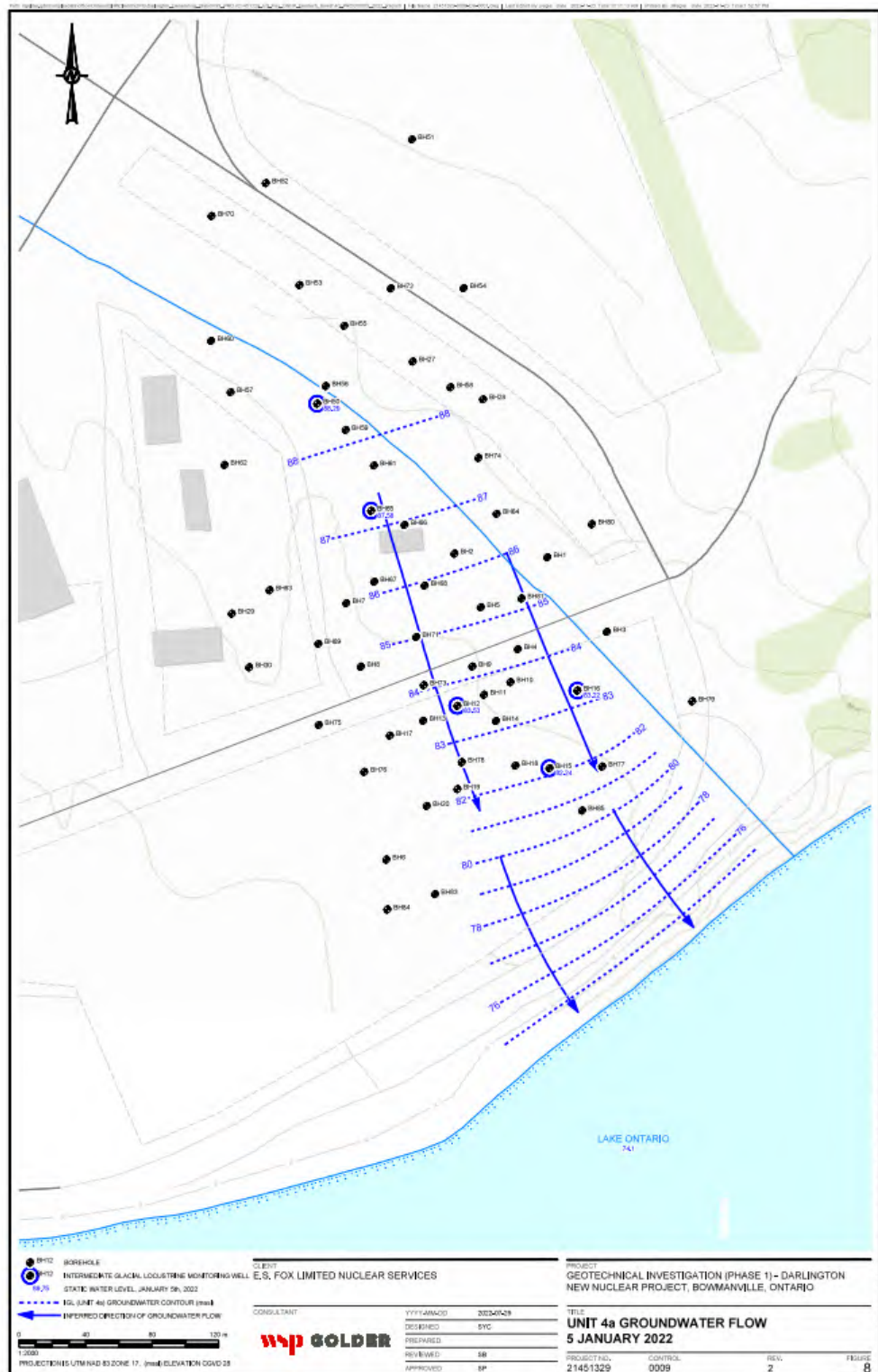


Figure 2.7.3.2-4: Unit 4a Groundwater Flow – Inter-glacial Deposits (Reference 2.7-39)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

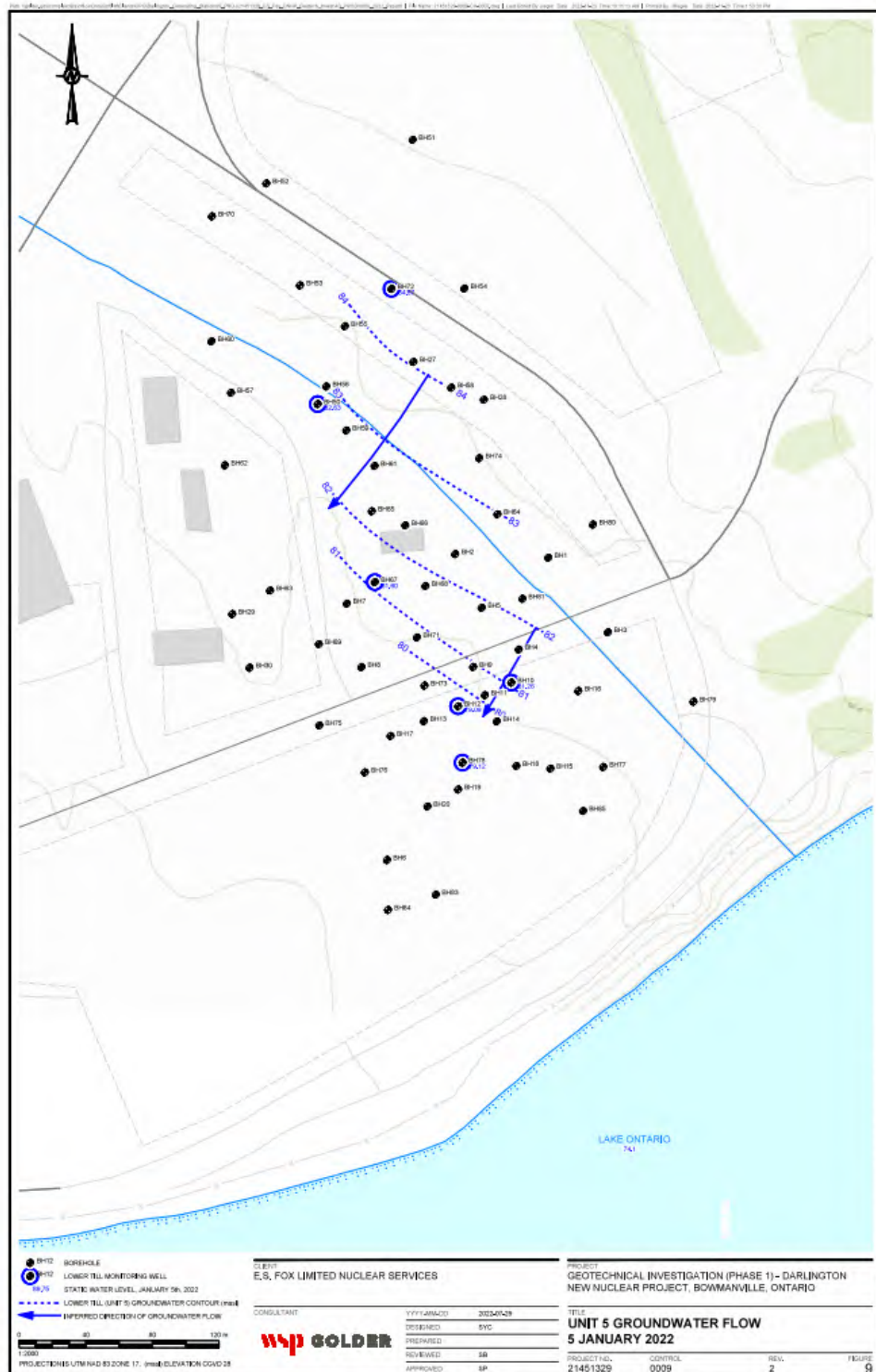


Figure 2.7.3.2-5: Unit 5 Groundwater Flow – Shallow Bedrock (Reference 2.7-39)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

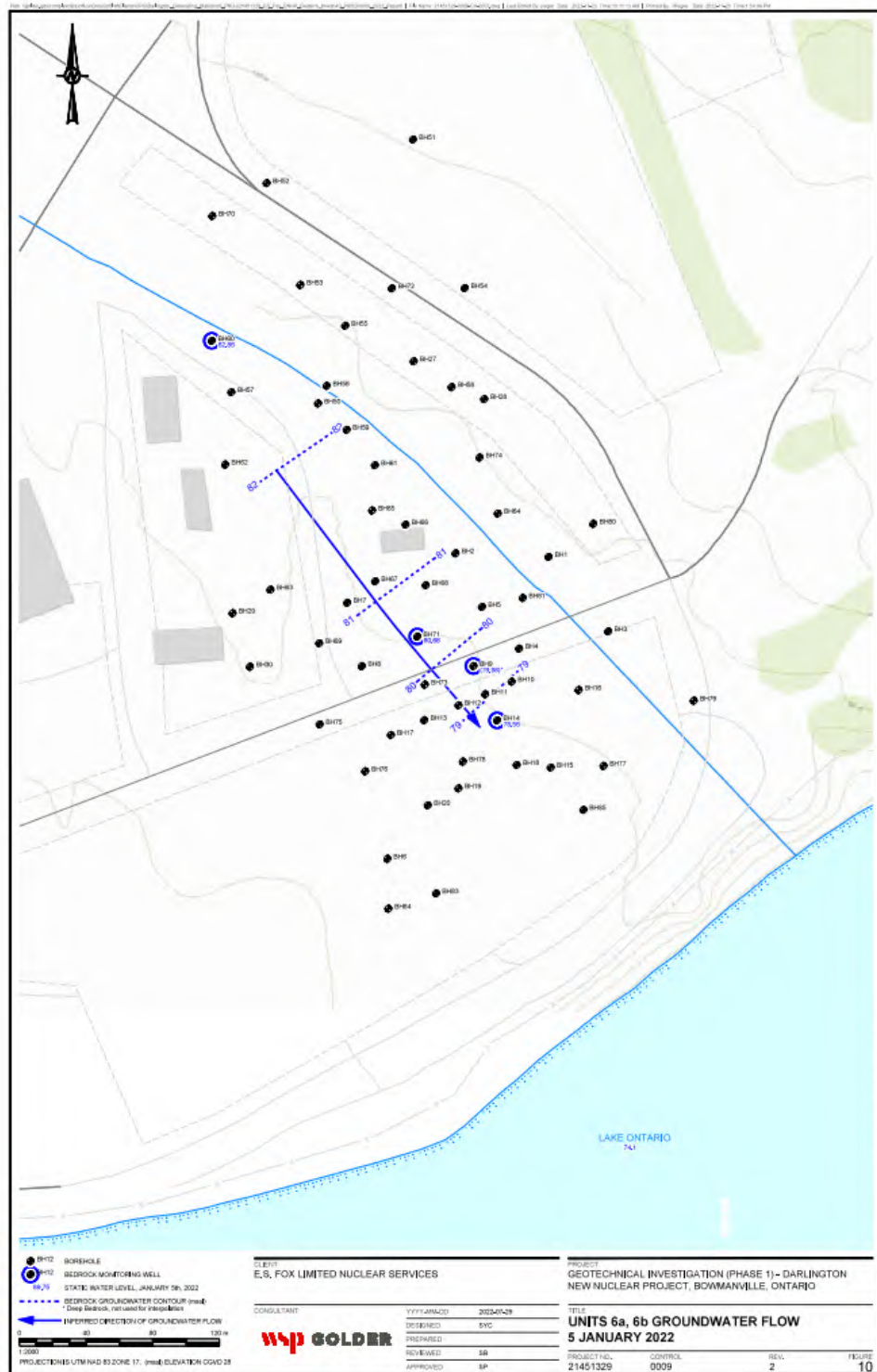


Figure 2.7.3.2-6: Units 6a - 6b Groundwater Flow – Shallow Bedrock (Reference 2.7-39)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

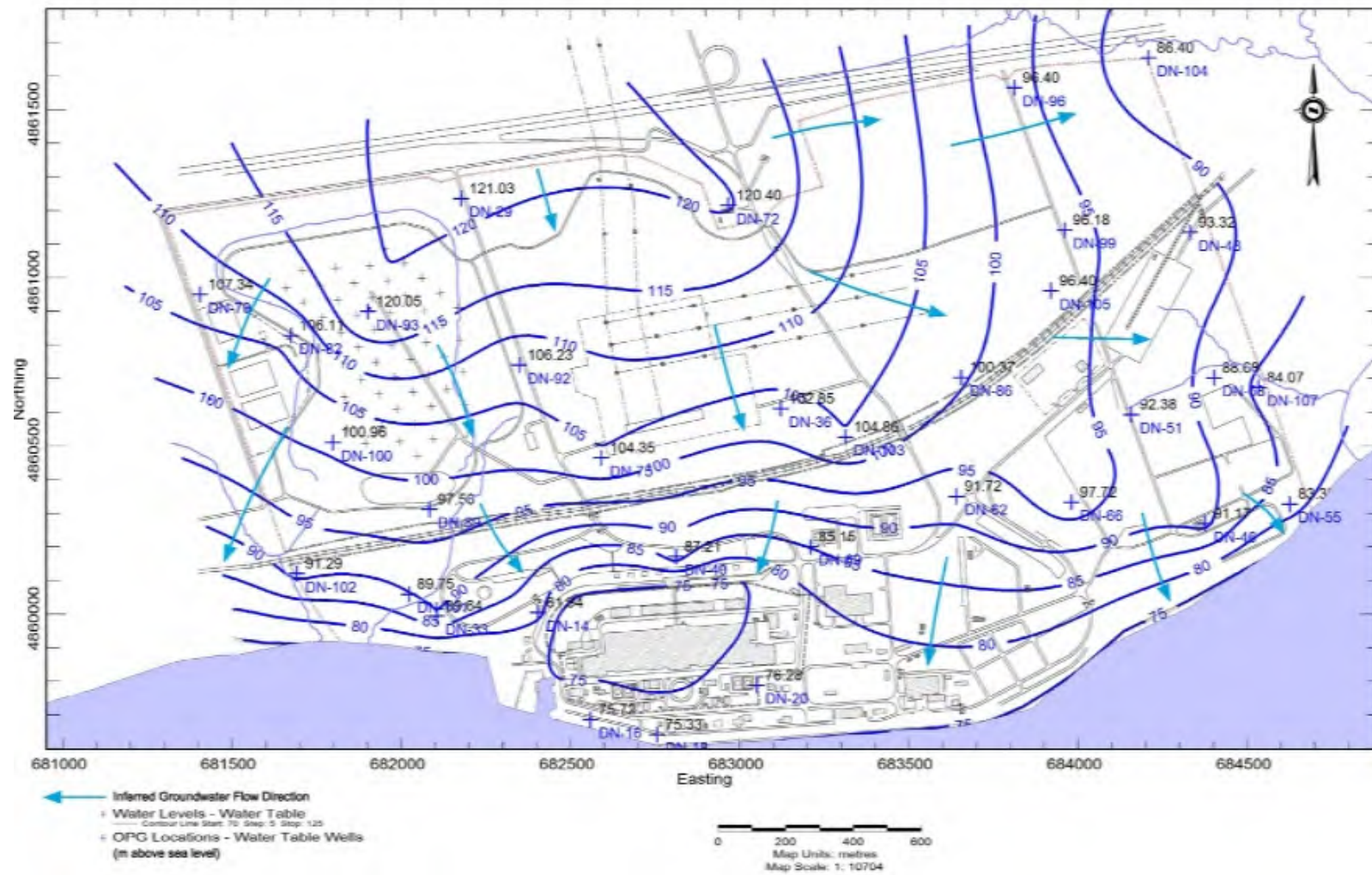


Figure 2.7.3.2-7: Regional Groundwater Levels – Shallow/Water Table (Reference 2.7-1)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

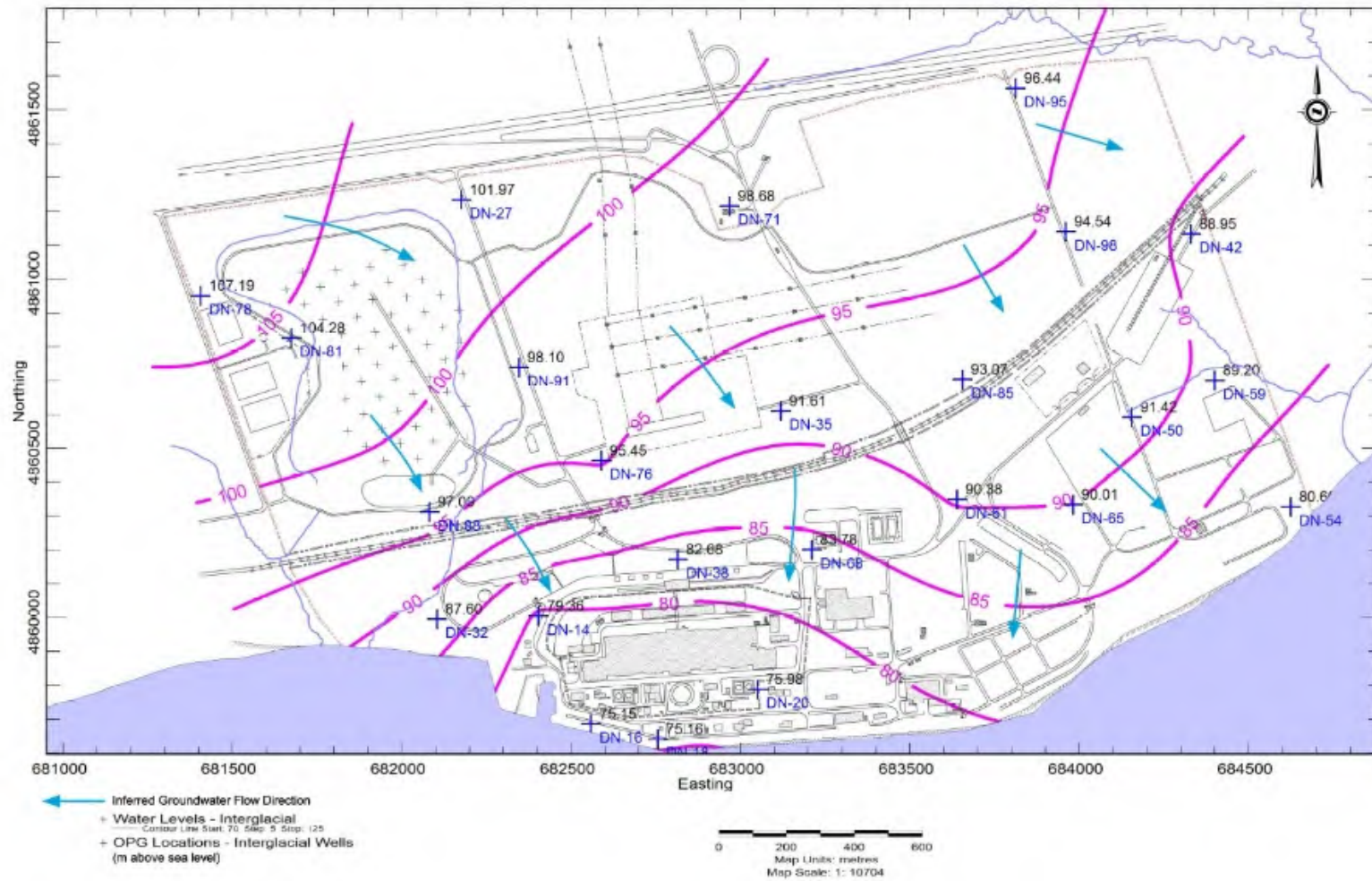


Figure 2.7.3.2-8: Regional Groundwater Flow – Interglacial Deposits (Reference 2.7-1)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

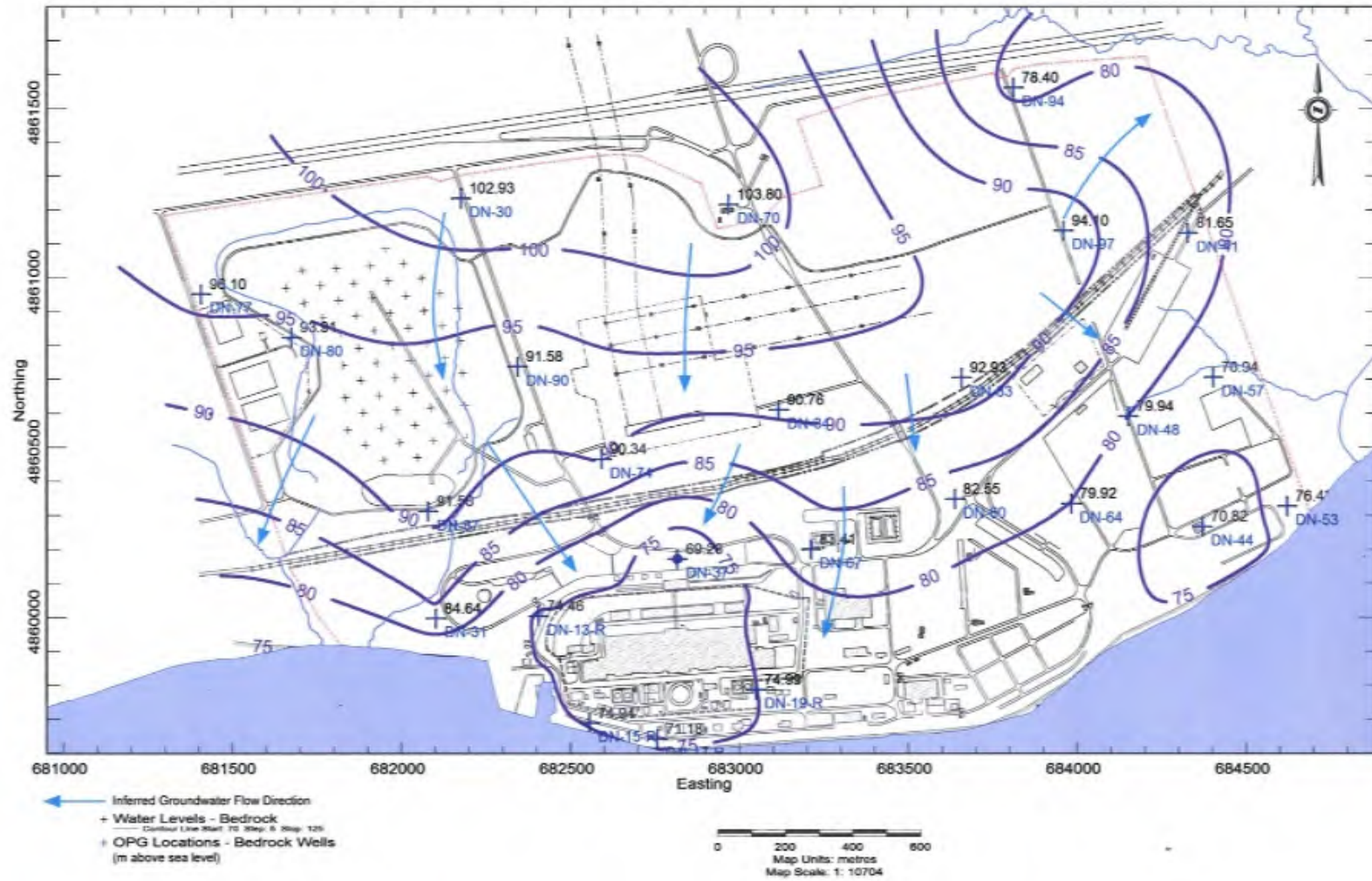


Figure 2.7.3.2-9: Regional Groundwater Flow – Shallow Bedrock (Reference 2.7-1)

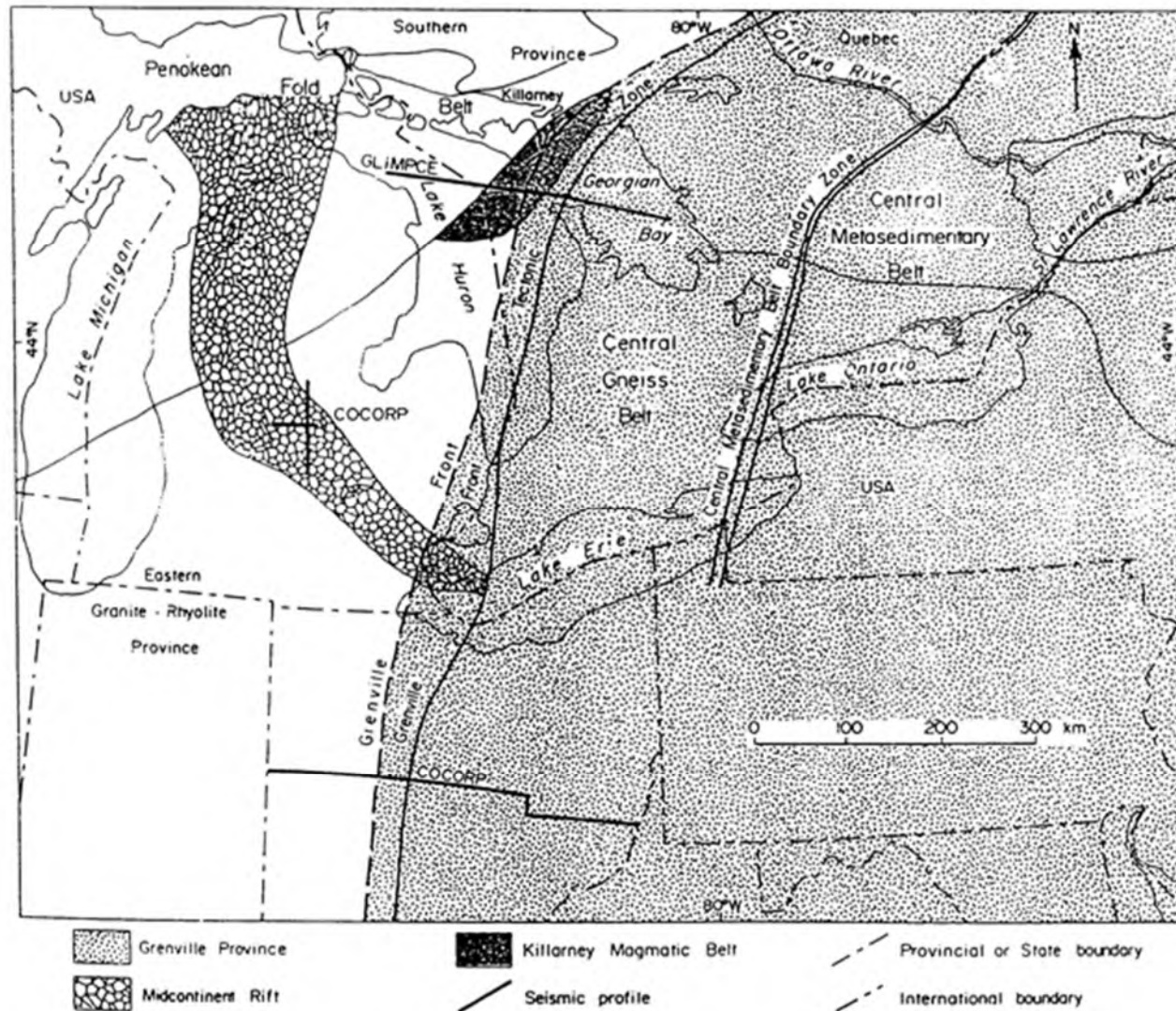
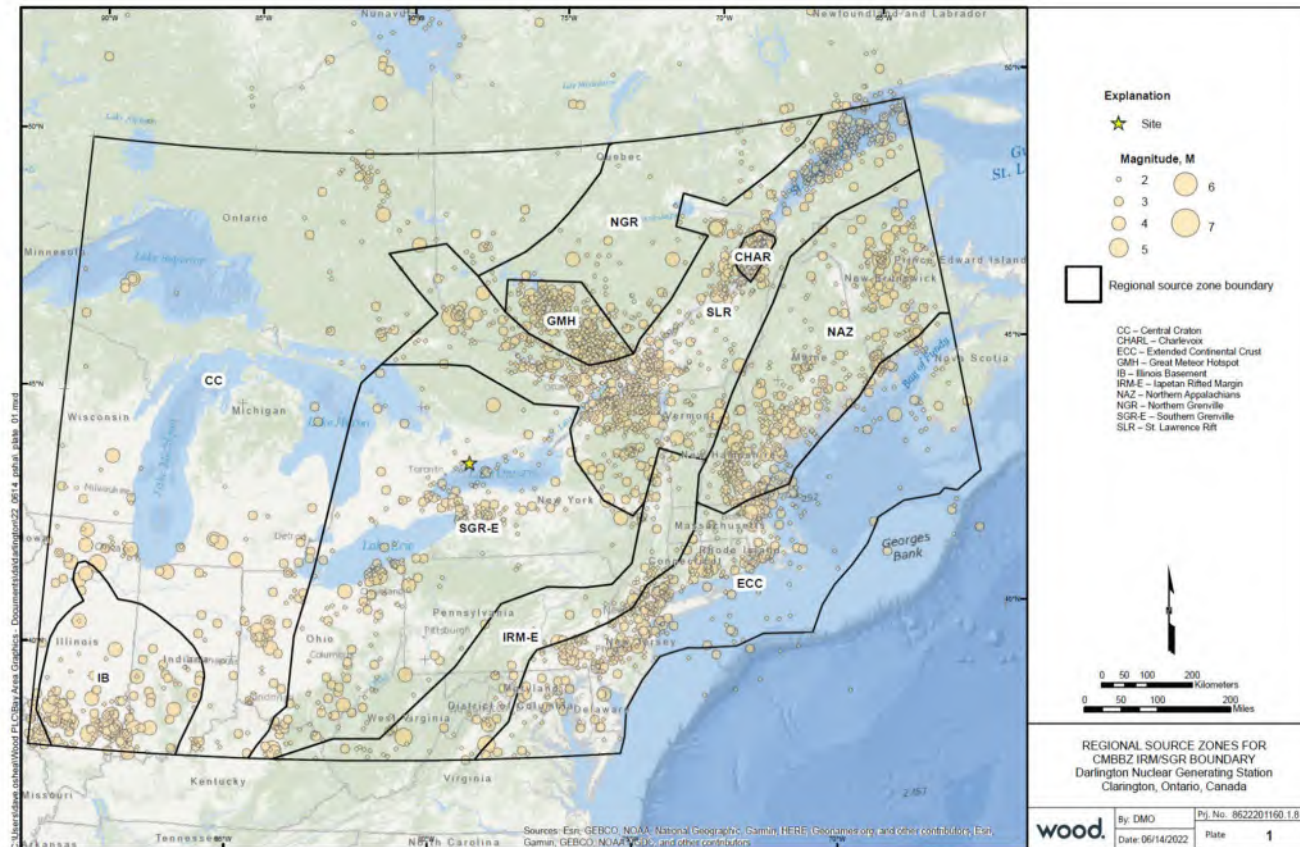


Figure 2.7.4.2-1: Principal Subdivisions of Precambrian Rocks in the Great Lakes Region (Reference 2.7-5)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

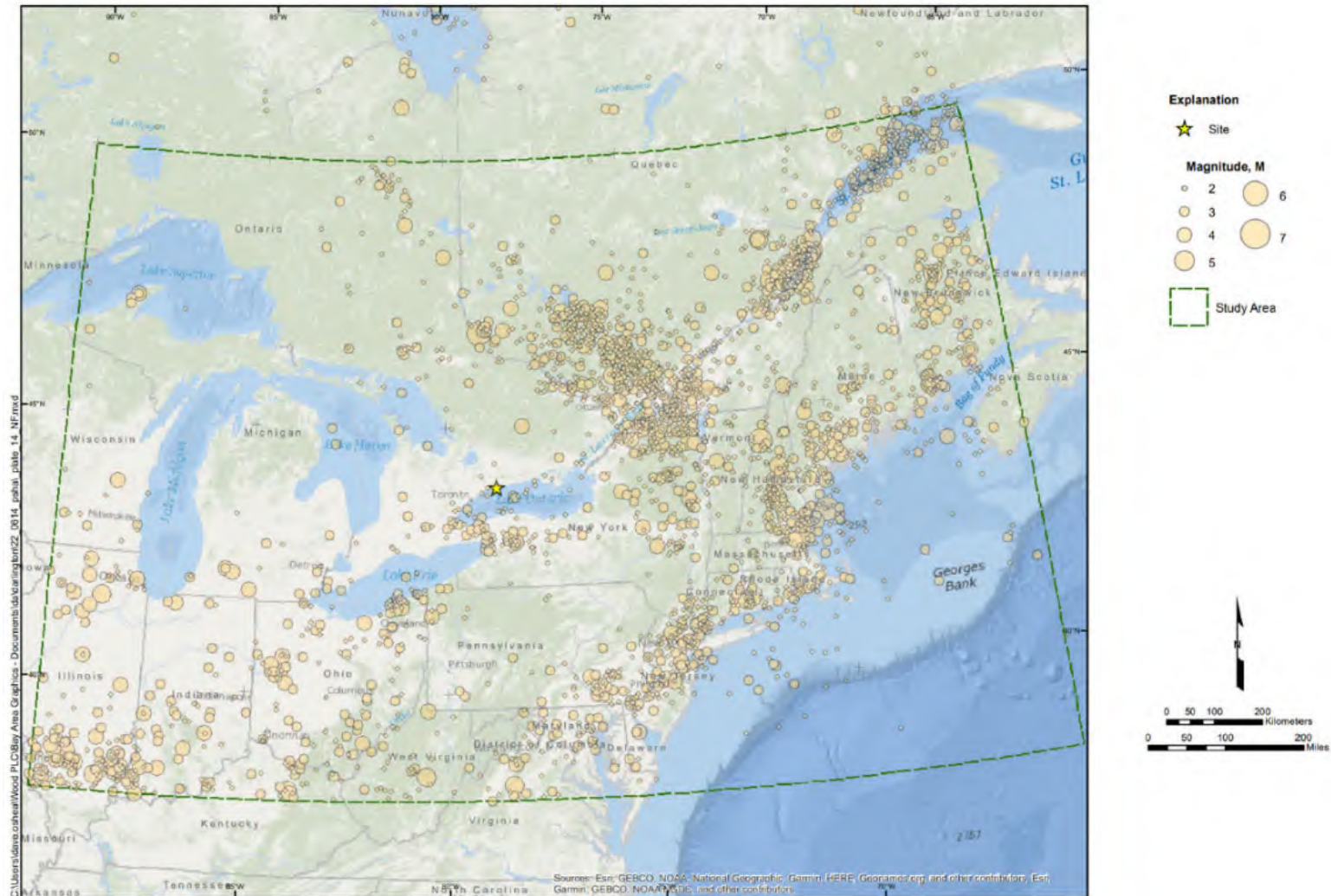


Abbreviations:

CC – Central Craton  
CHAR – Charlevoix  
ECC – Extended Continental Crust  
GMH – Great Meteor Hotspot  
IB – Illinois Basement

IRM-E – Iapetan Rifted Margin  
NAZ – Northern Appalachians  
NGR – Northern Grenville  
SGR-E – Southern Grenville  
SLR – St. Lawrence Rift

**Figure 2.7.4.2-2: Regional Source Zones for IRM/SGR Boundary Eastern Boundary (Reference 2.7-41)**



**Figure 2.7.4.3-1: Map of Independent Earthquakes in the Updated Earthquake Catalogue for the Study Region (Reference 2.7-41)**

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

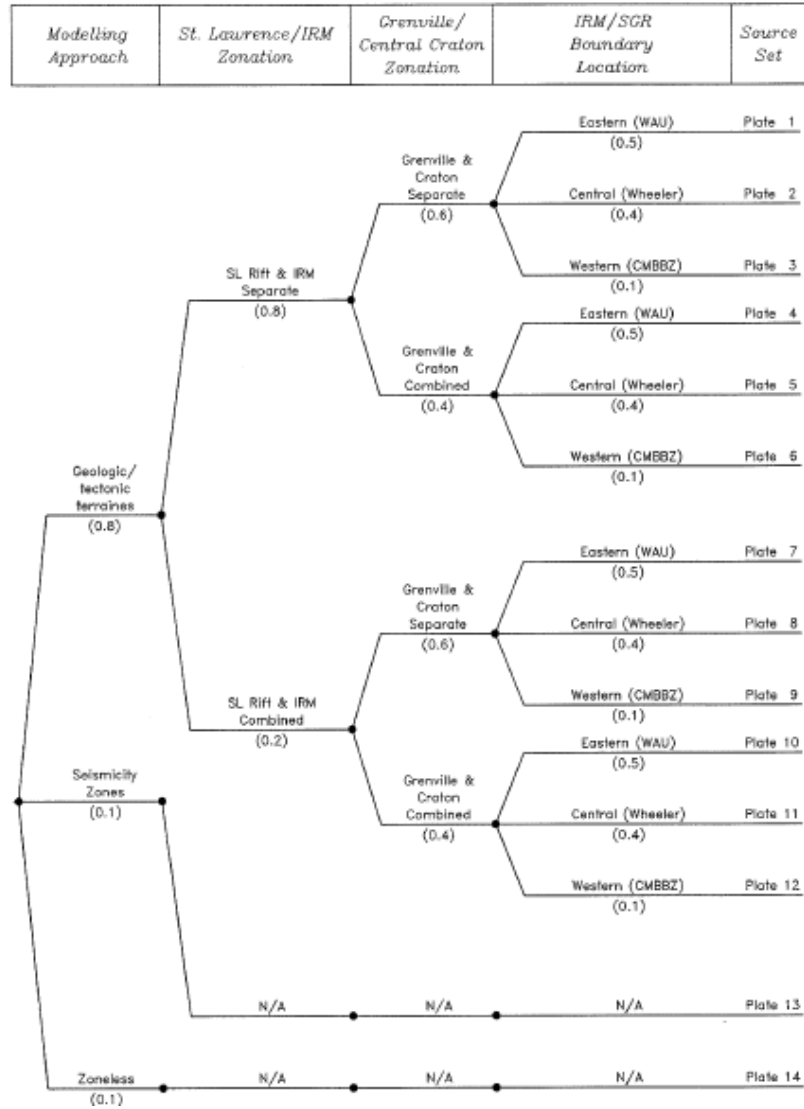


Figure 2.7.4.4-1: Logic Tree for Distributed Seismicity Sources (Reference 2.7- 42)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

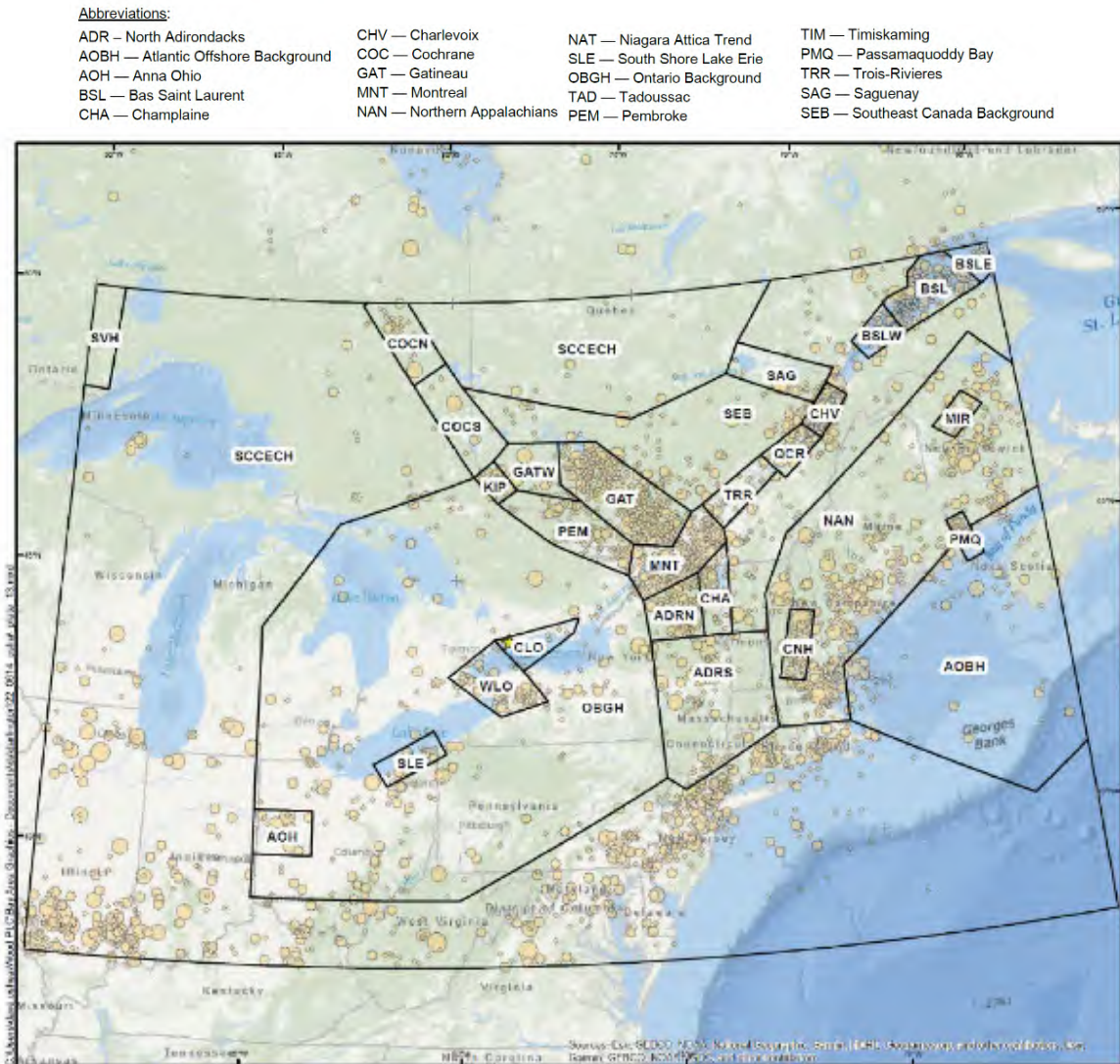


Figure 2.7.4.4-2: Regional Seismicity Source Zones from the Geological Survey of Canada's Sixth Generation H Model  
(Plate 13 in Reference 2.7-41)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

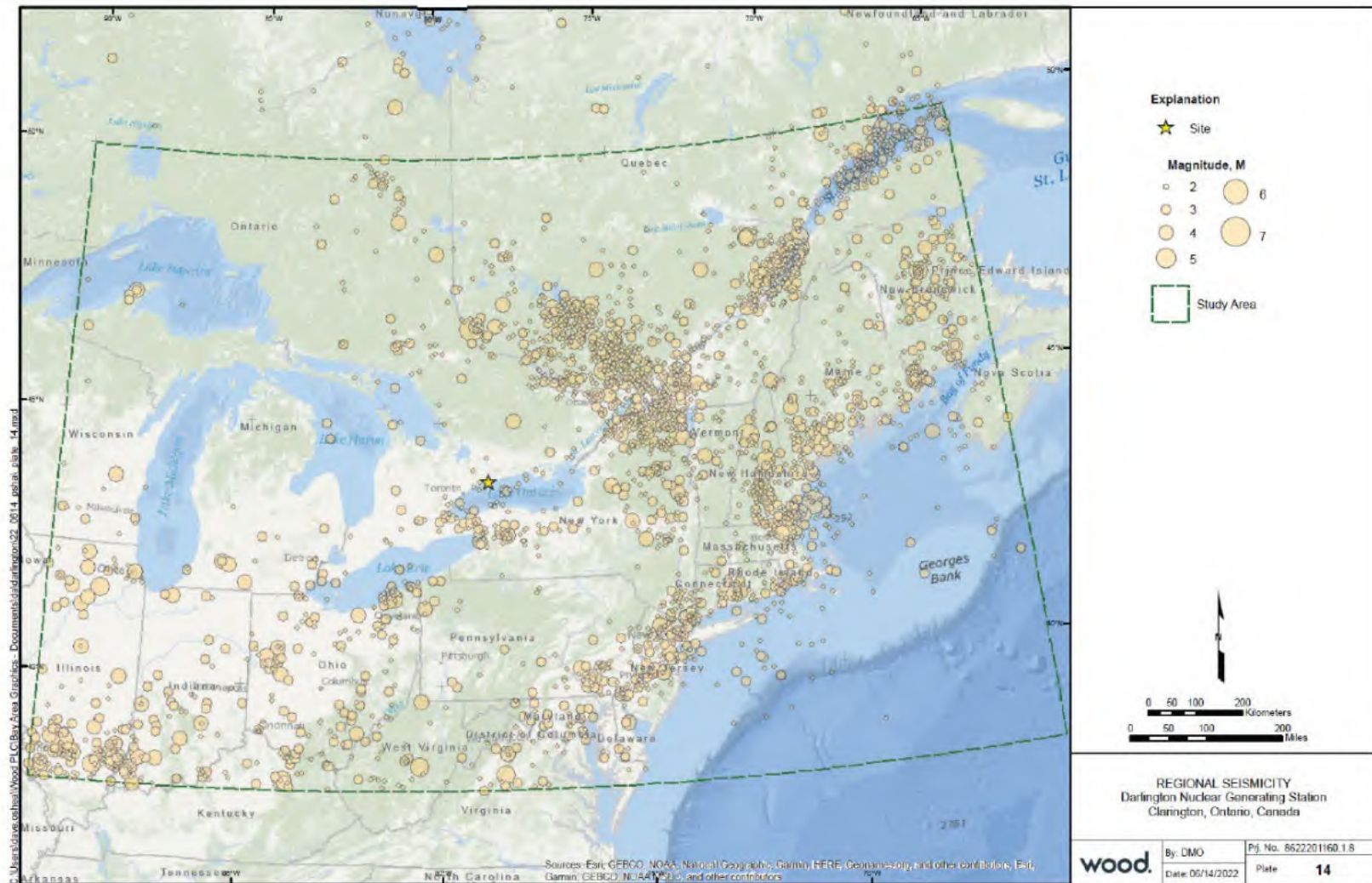
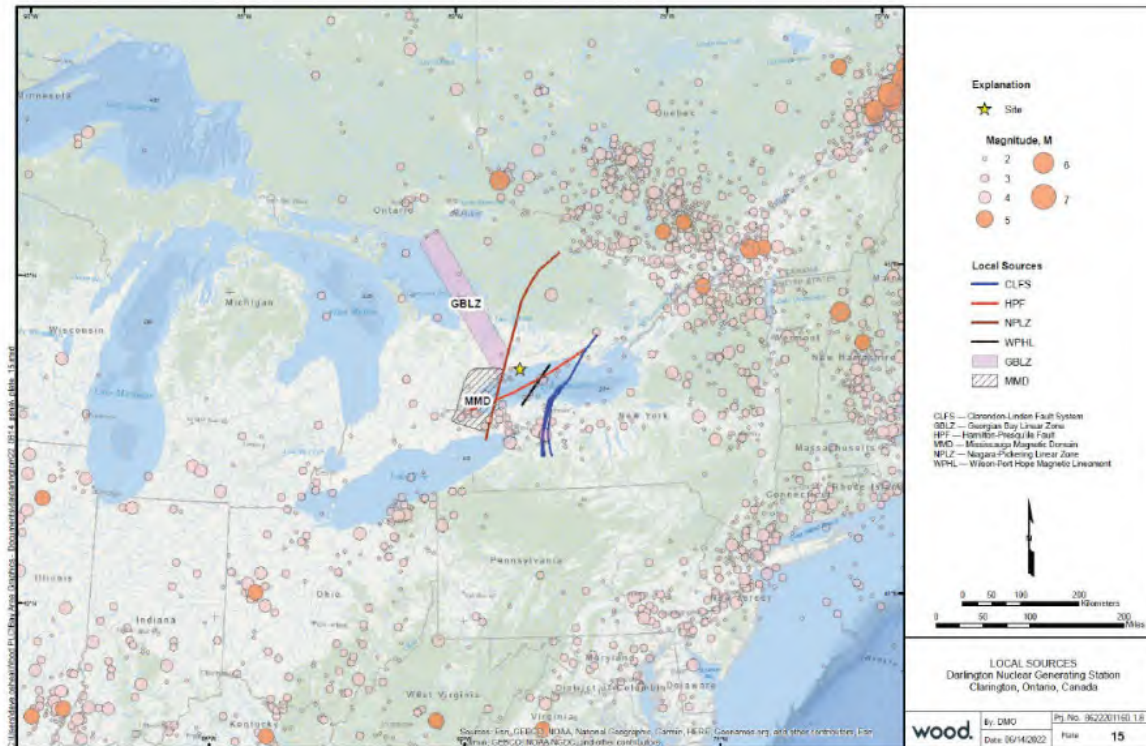


Figure 2.7.4.4-3: Regional Seismicity for Zoneless Model (Plate 14 in Reference 2.7-41)



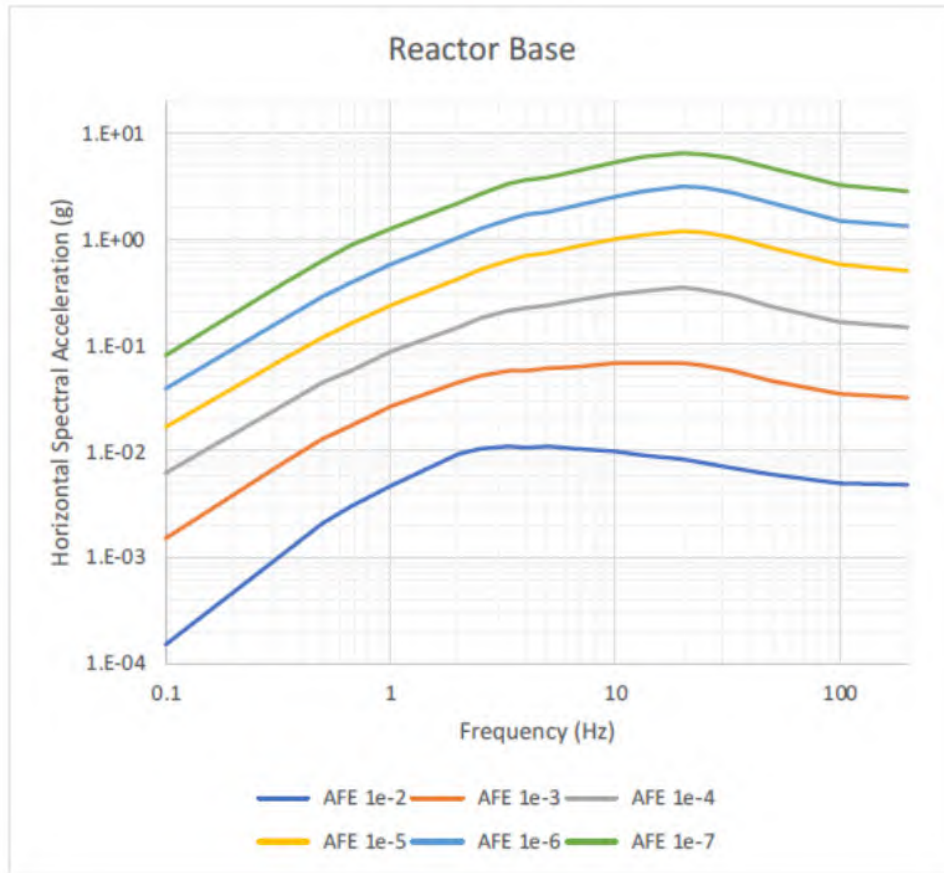
NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



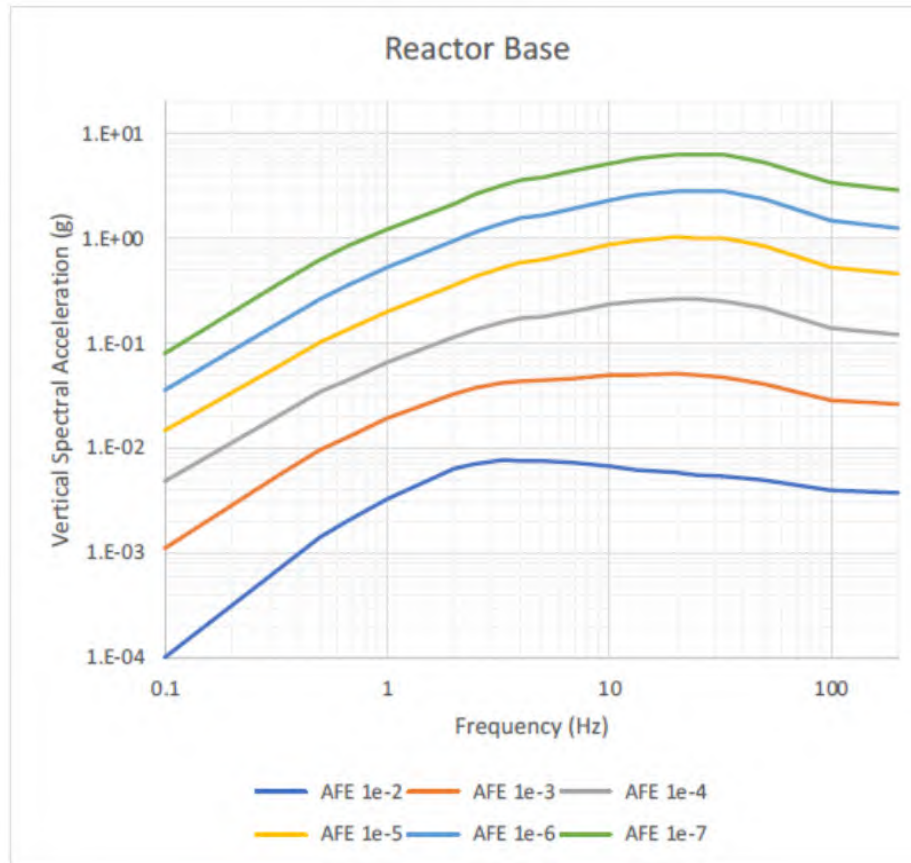
Abbreviations:

CLFS — Clarendon-Linden Fault System  
GBLZ — Georgian Bay Linear Zone  
HPF — Hamilton-Presqu'ile Fault  
MMD — Mississauga Magnetic Domain  
NPLZ — Niagara-Pickering Linear Zone  
WPHL — Wilson-Port Hope Magnetic Lineament

**Figure 2.7.4.4-4: Local Source Zones (Plate 15 in Reference 2.7-41)**

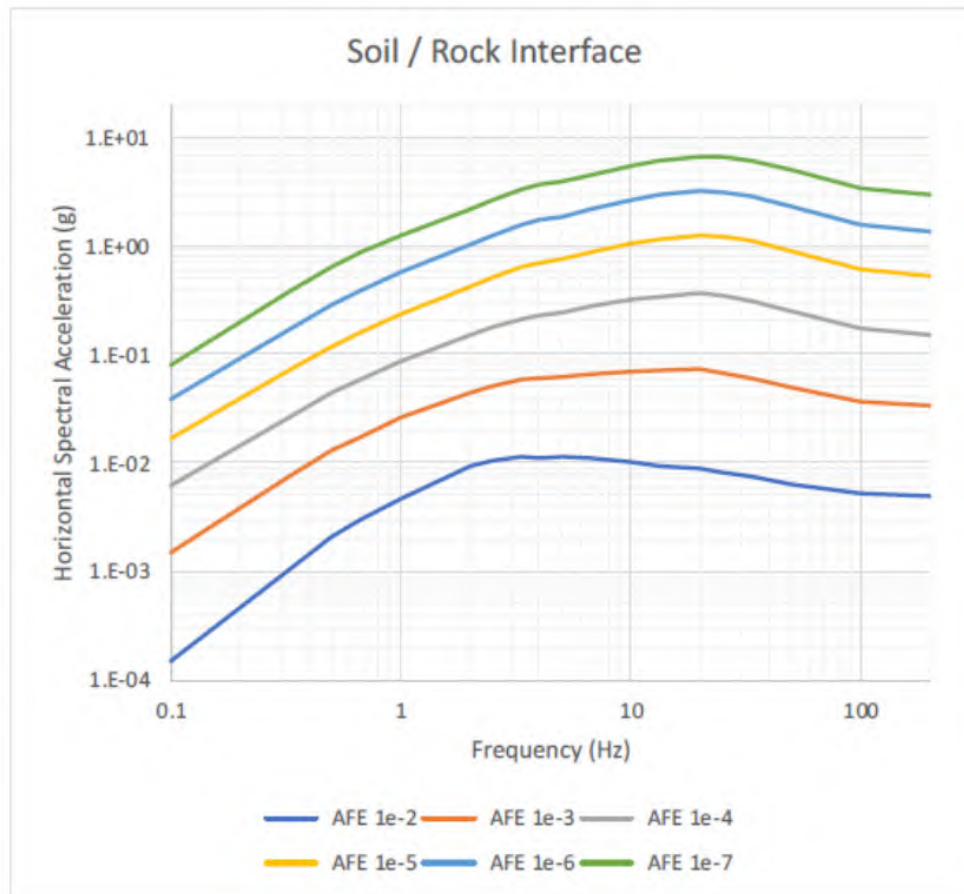


**Figure 2.7.4.6-1: Horizontal UHRS at Elevation 52.93 m Based on Mean Hazard  
(Reference 2.7-41)**

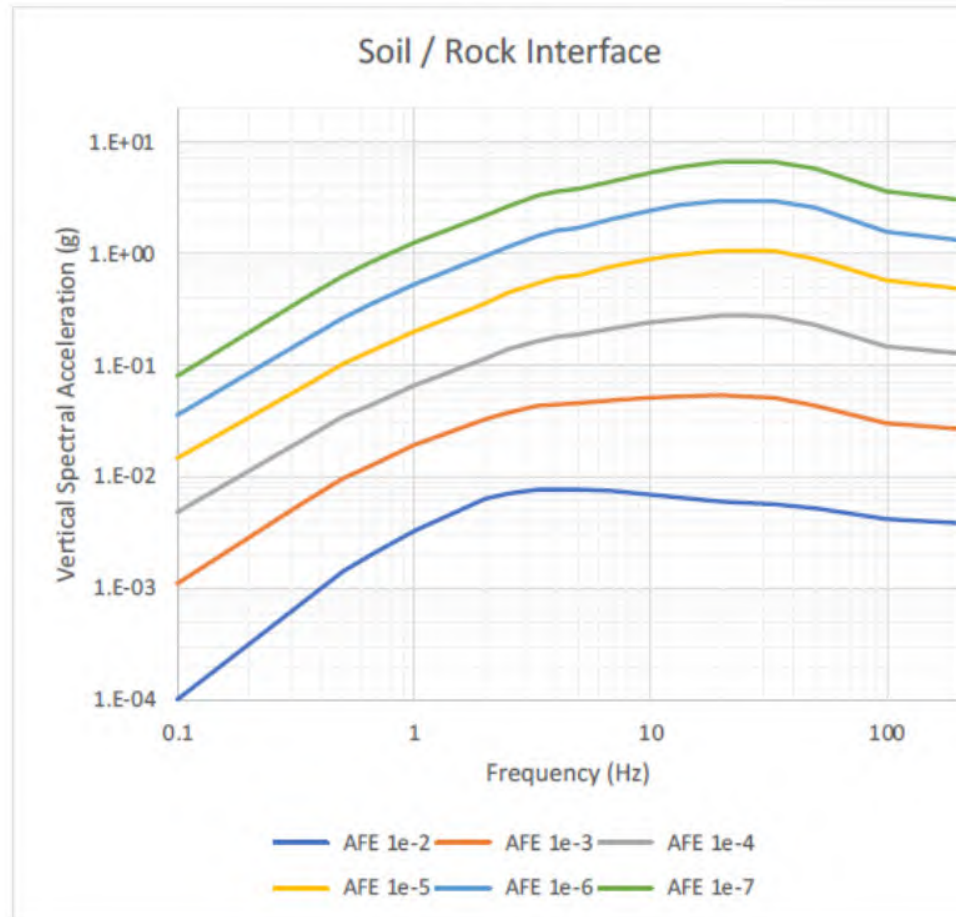


**Figure 2.7.4.6-2: Vertical UHRS at Elevation 52.93 m Based on Mean Hazard  
(Reference 2.7-41)**

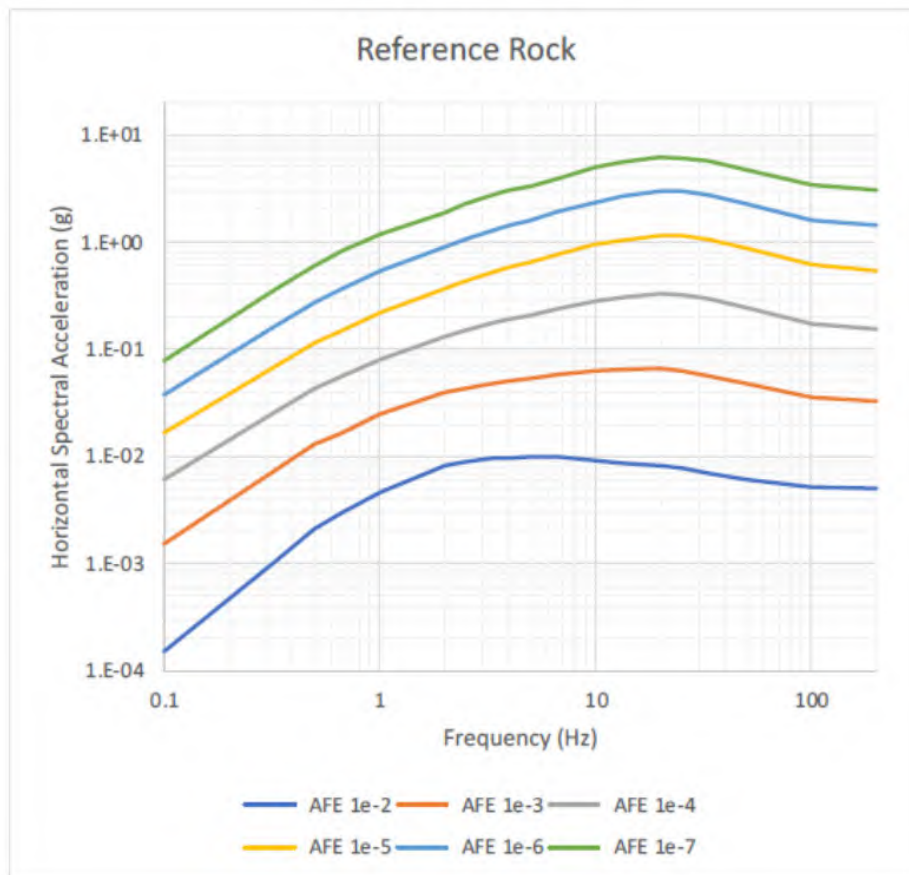




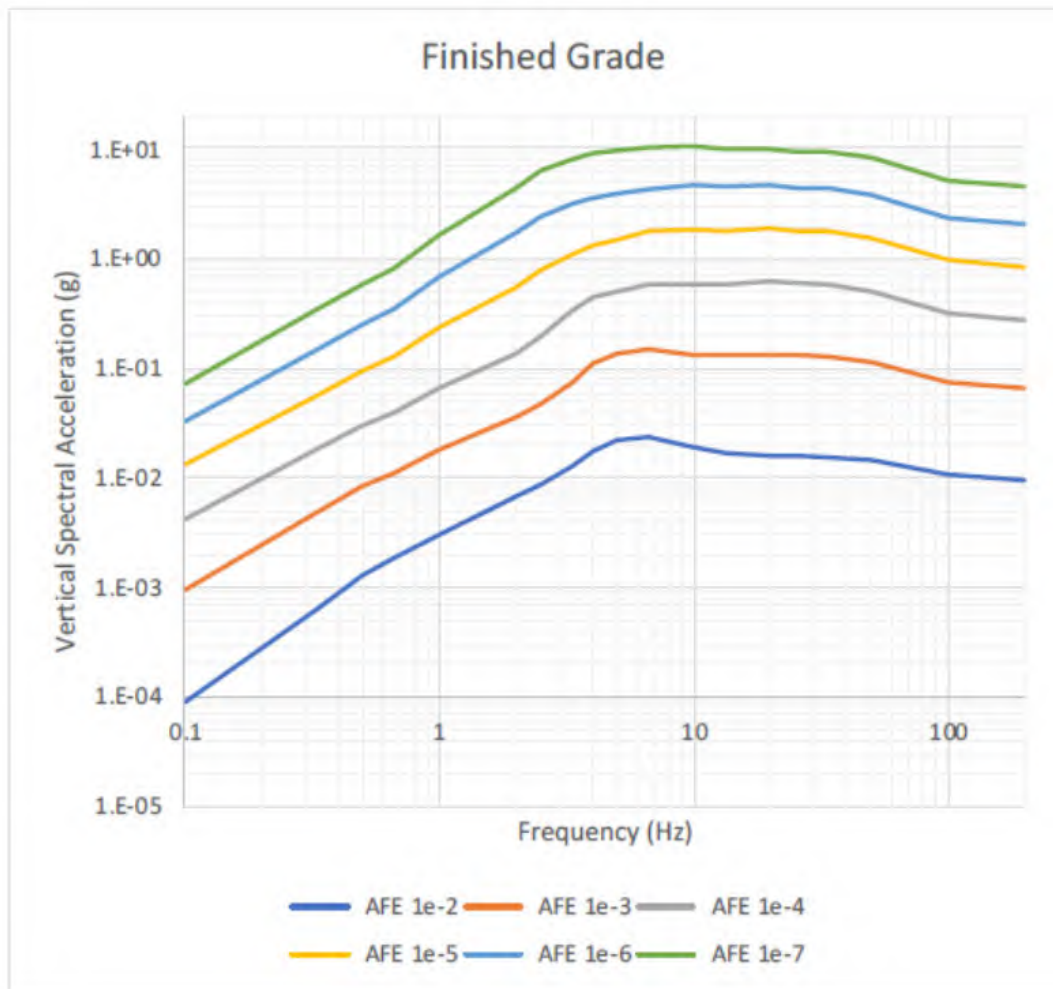
**Figure 2.7.4.6-3: Horizontal UHRS at Elevation 64 m Based on Mean Hazard (Reference 2.7-41)**



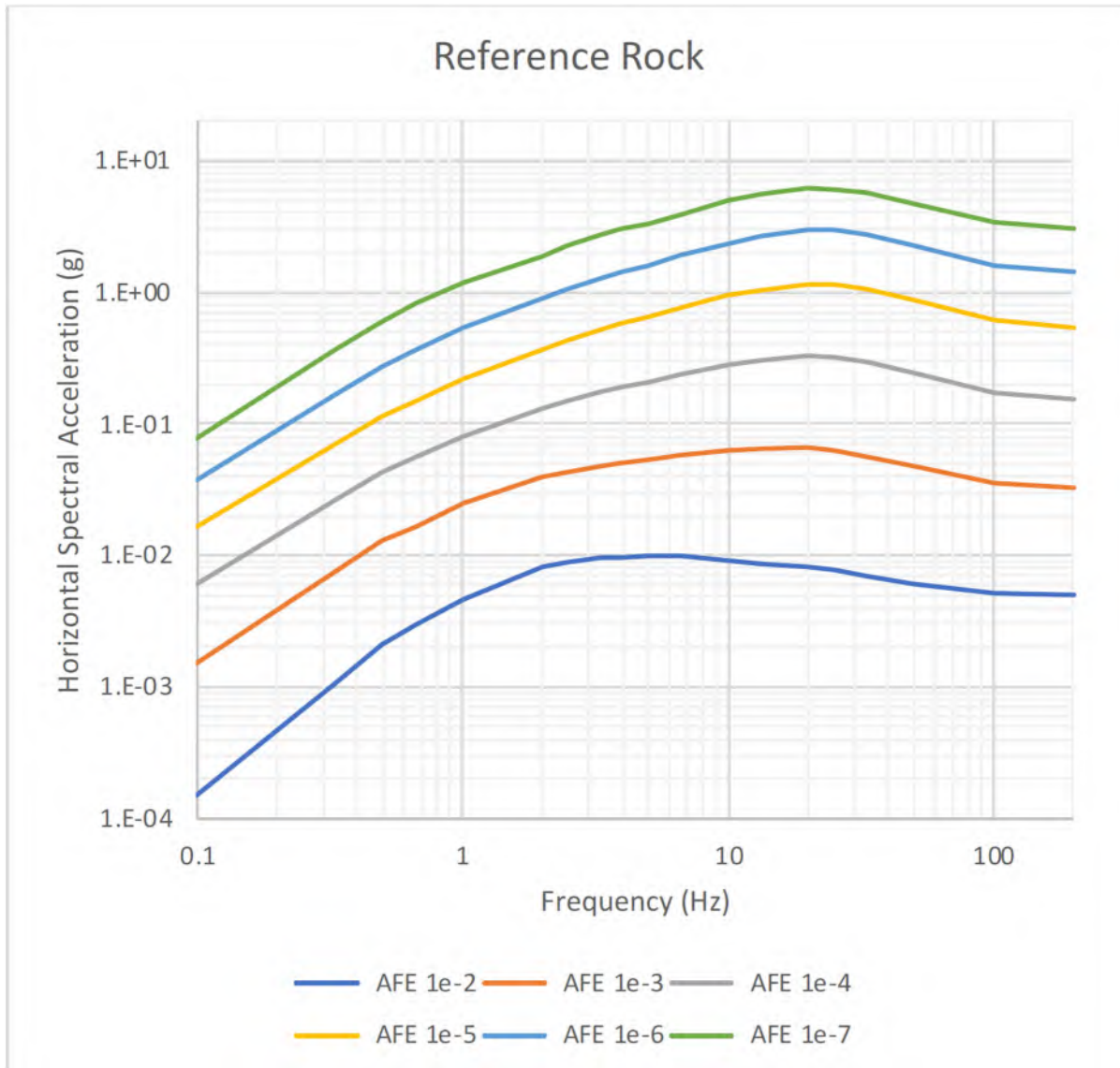
**Figure 2.7.4.6-4: Vertical UHRS at Elevation 64 m Based on Mean Hazard  
(Reference 2.7-41)**



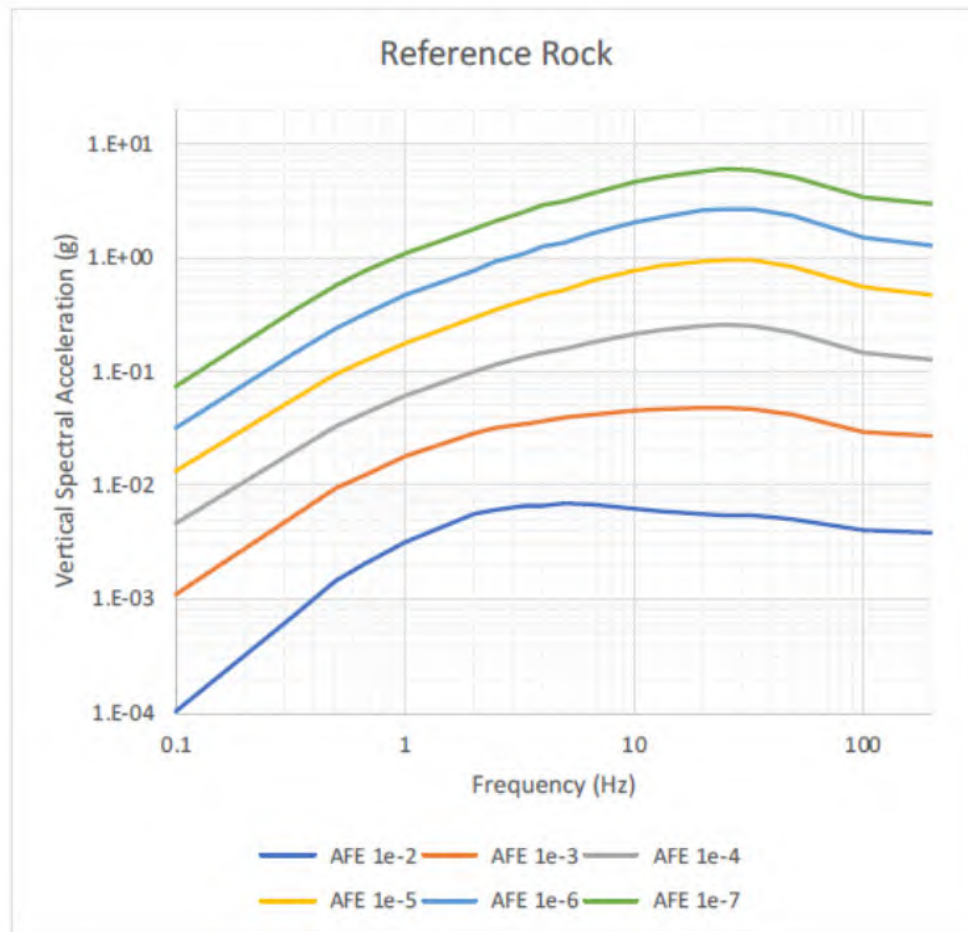
**Figure 2.7.4.6-5: Horizontal UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)**



**Figure 2.7.4.6-6: Vertical UHS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)**



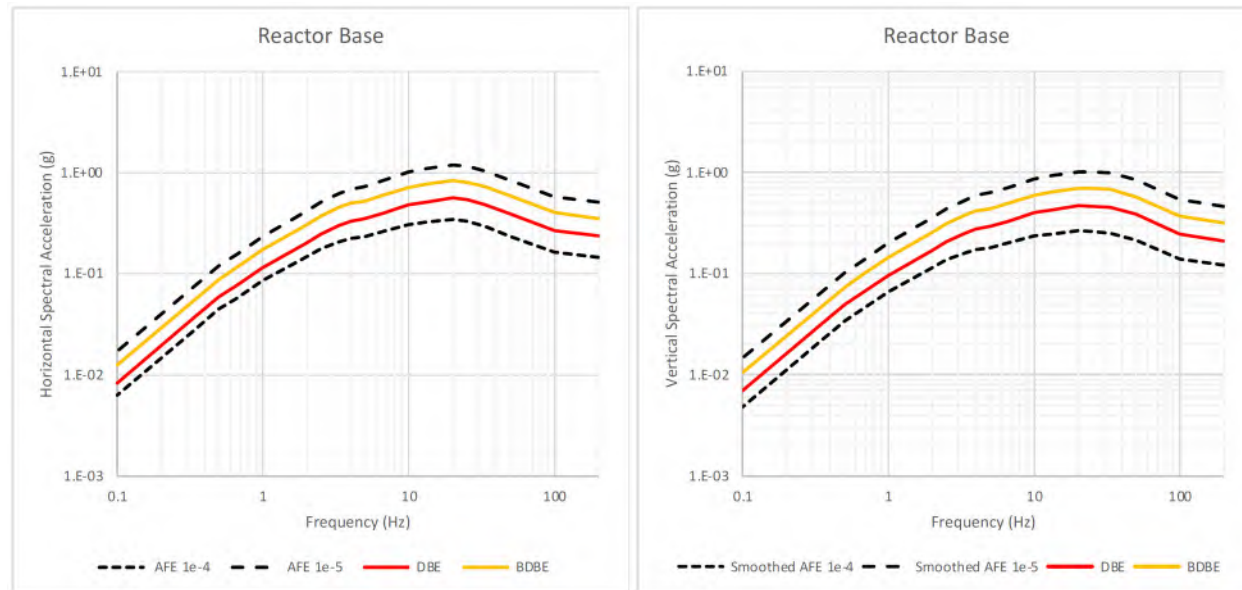
**Figure 2.7.4.6-7: Horizontal UHRS for Reference Rock Based on Mean Hazard (Reference 2.7-41)**



**Figure 2.7.4.6-8: Vertical UHRS for Reference Rock Based on Mean Hazard  
(Reference 2.7-41)**



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 2.7.4.6-9: Initial Horizontal DBE and BDBE Spectra for Reactor Base (Elevation 52.93 m)  
(Reference 2.7-41)**

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

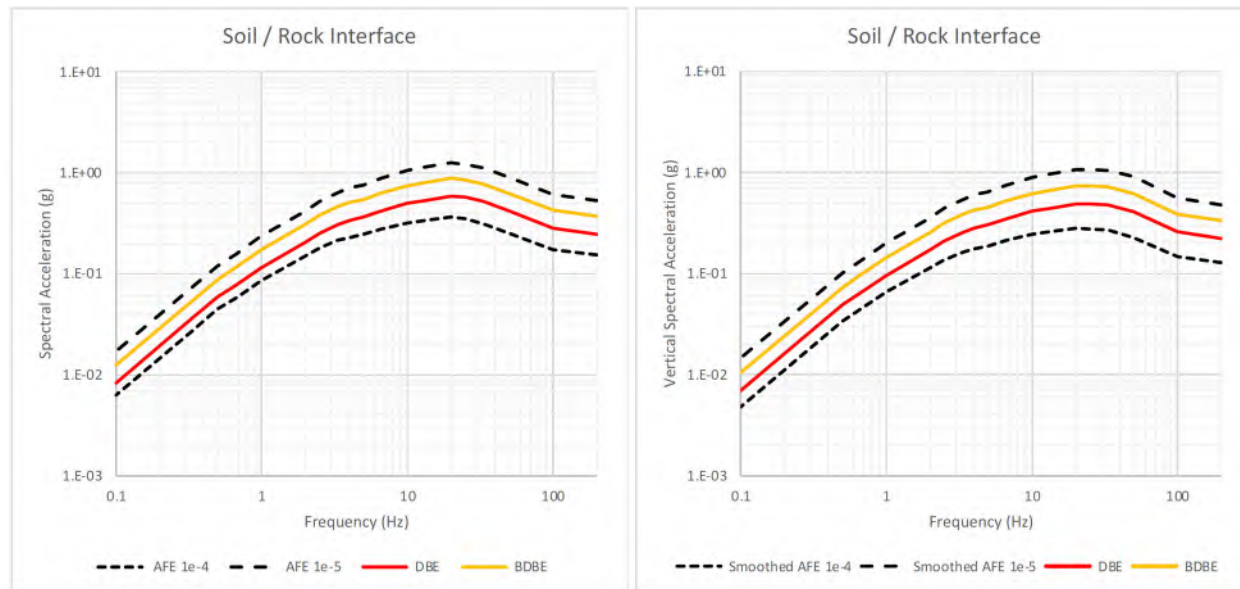
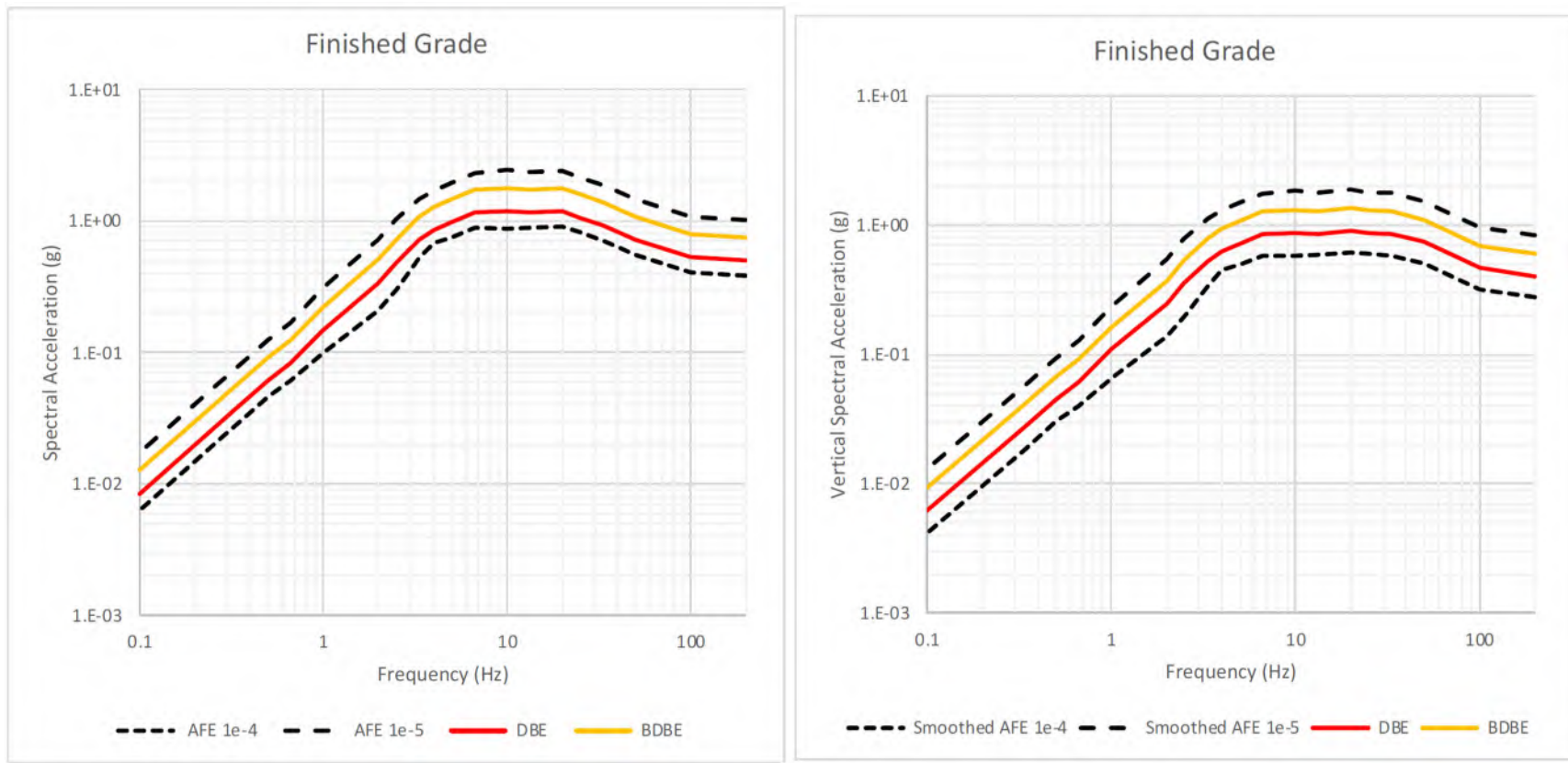


Figure 2.7.4.6-10: Initial Horizontal DBE and BDBE Spectra for Soil-Rock Interface (Elevation 64 m) (Reference 2.7-41)





**Figure 2.7.4.6-11: Initial Horizontal DBE and BDBE Spectra for Finished Grade (Elevation 88 m) (Reference 2.7-41)**

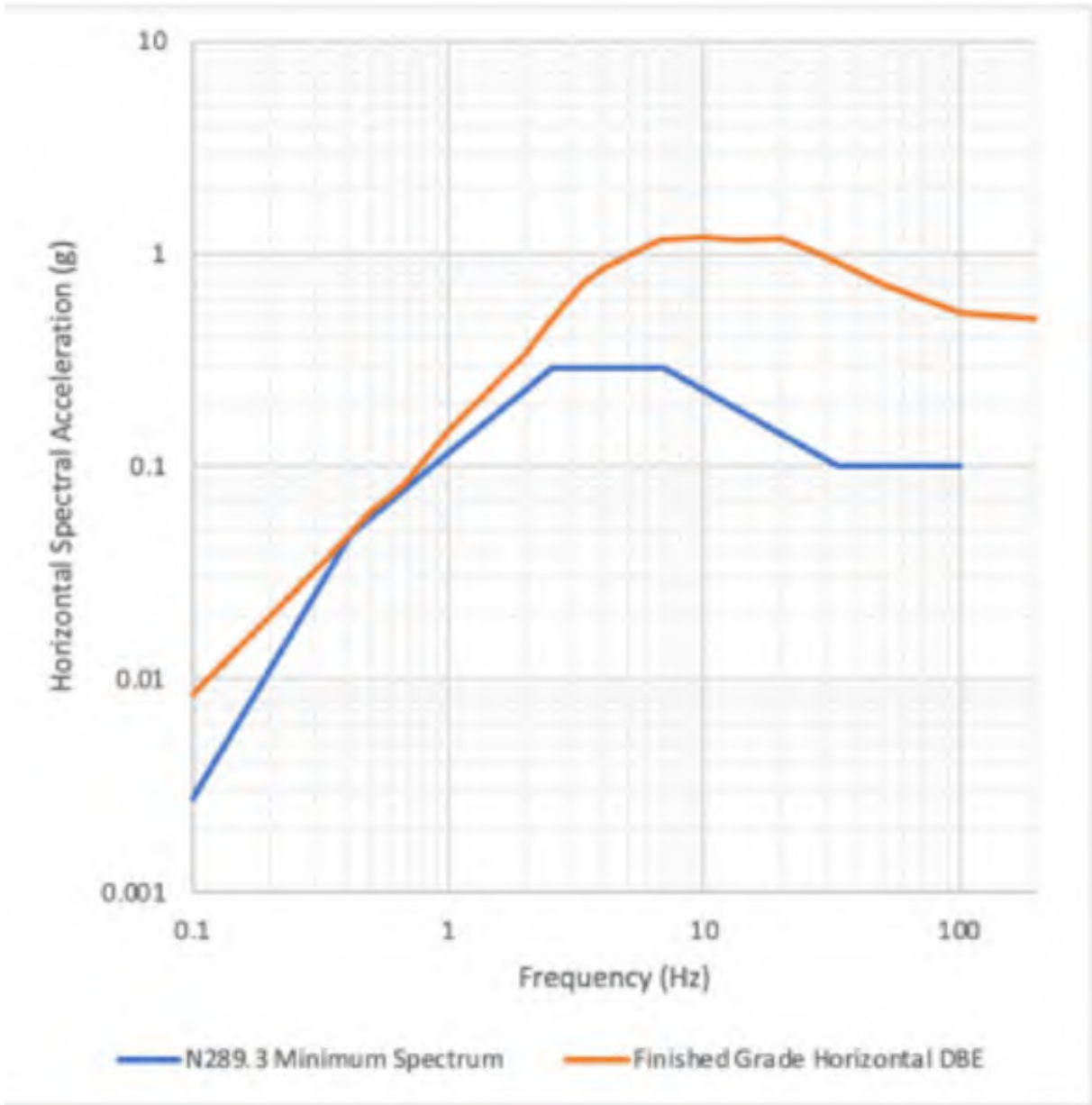


Figure 2.7.4.6-12: Comparison of Finished Grade Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)

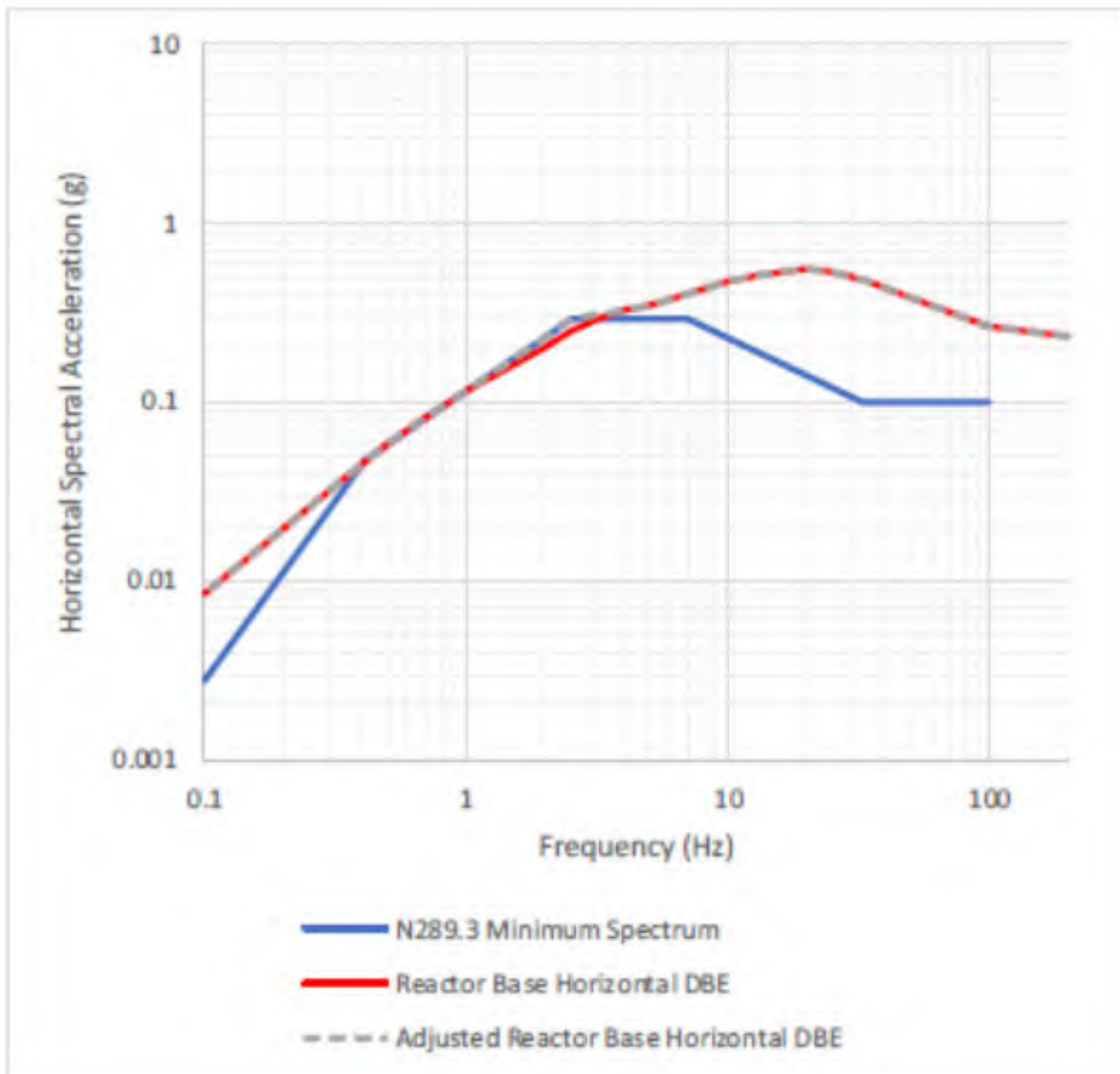


Figure 2.7.4.6-13: Comparison of Reactor Building Base Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

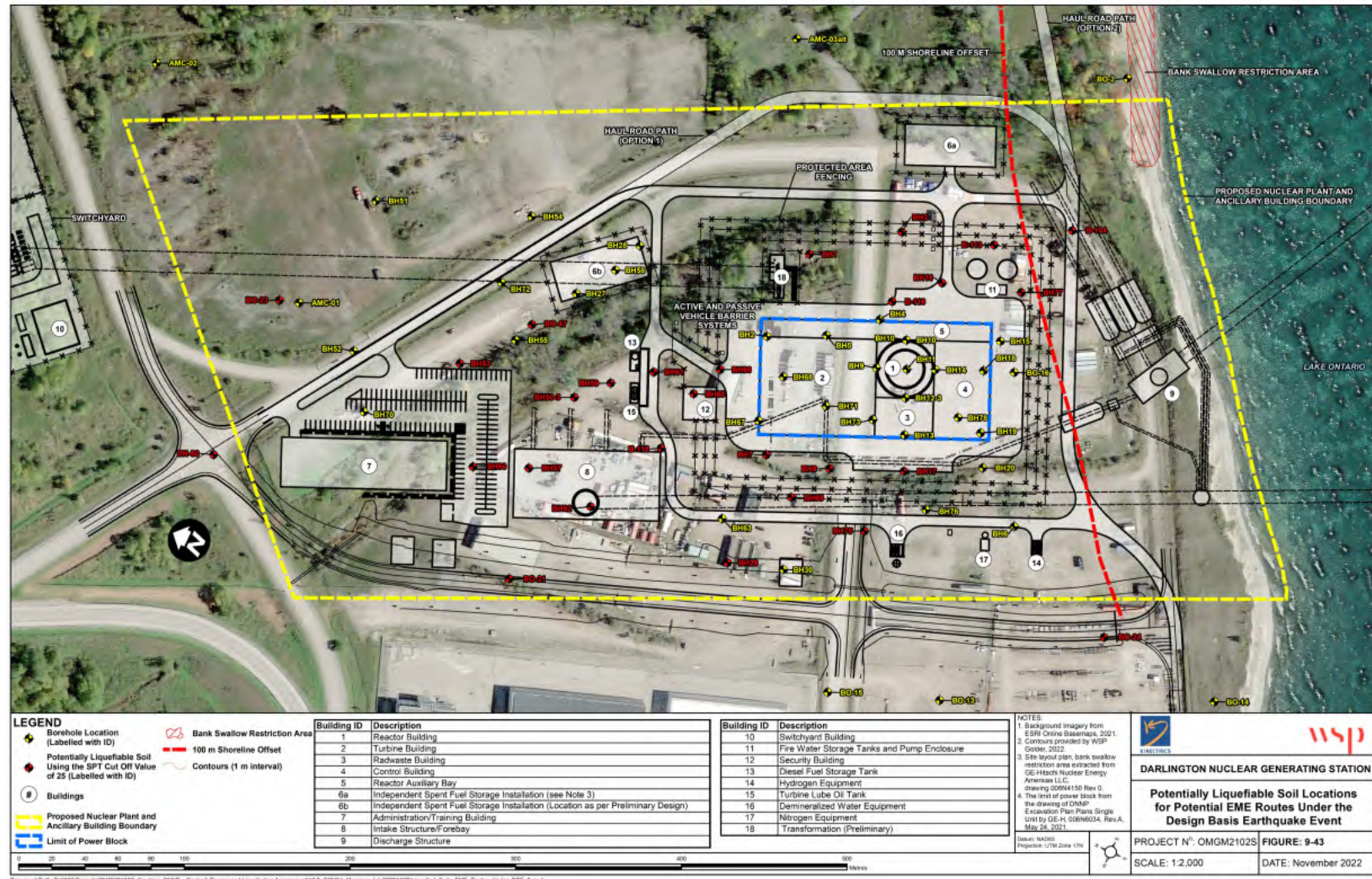


Figure 2.7.4.7-1: Locations of Boreholes Potentially to Liquefaction Under the DBE Event (Reference 2.7-42)



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

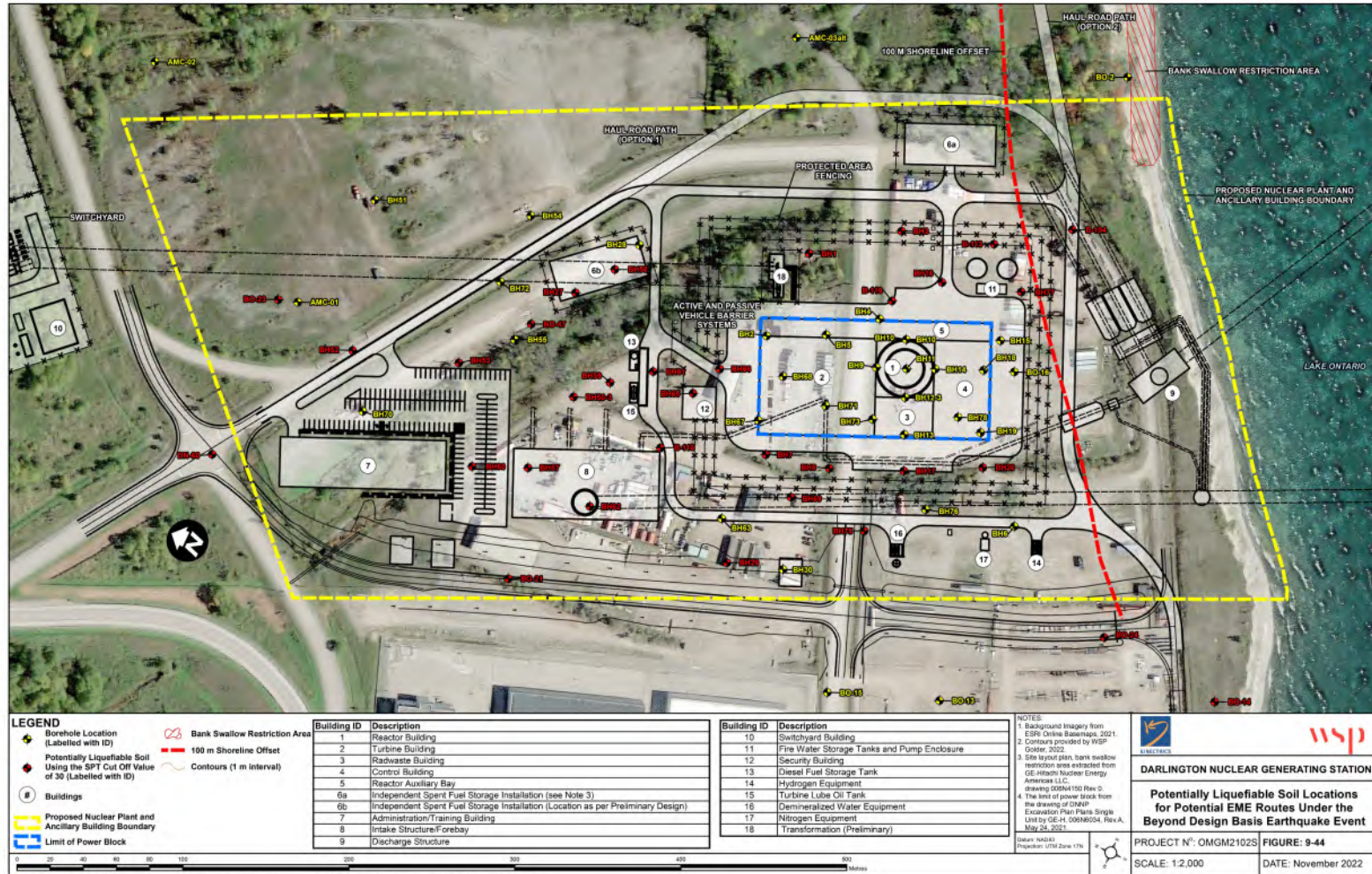


Figure 2.7.4.7-2: Locations of Boreholes Potentially to Liquefaction Under the BDBE Event (Reference 2.7-42)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

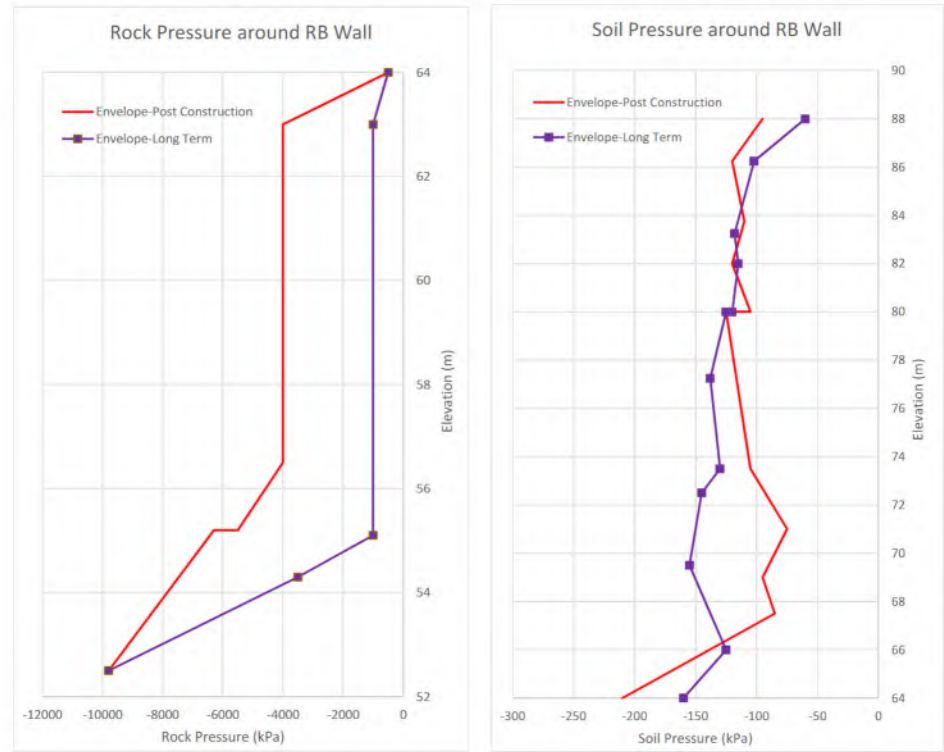


Figure 2.7.5.1.3-1: Rock and Soil Pressures Around RB Wall (Reference 2.7-38)

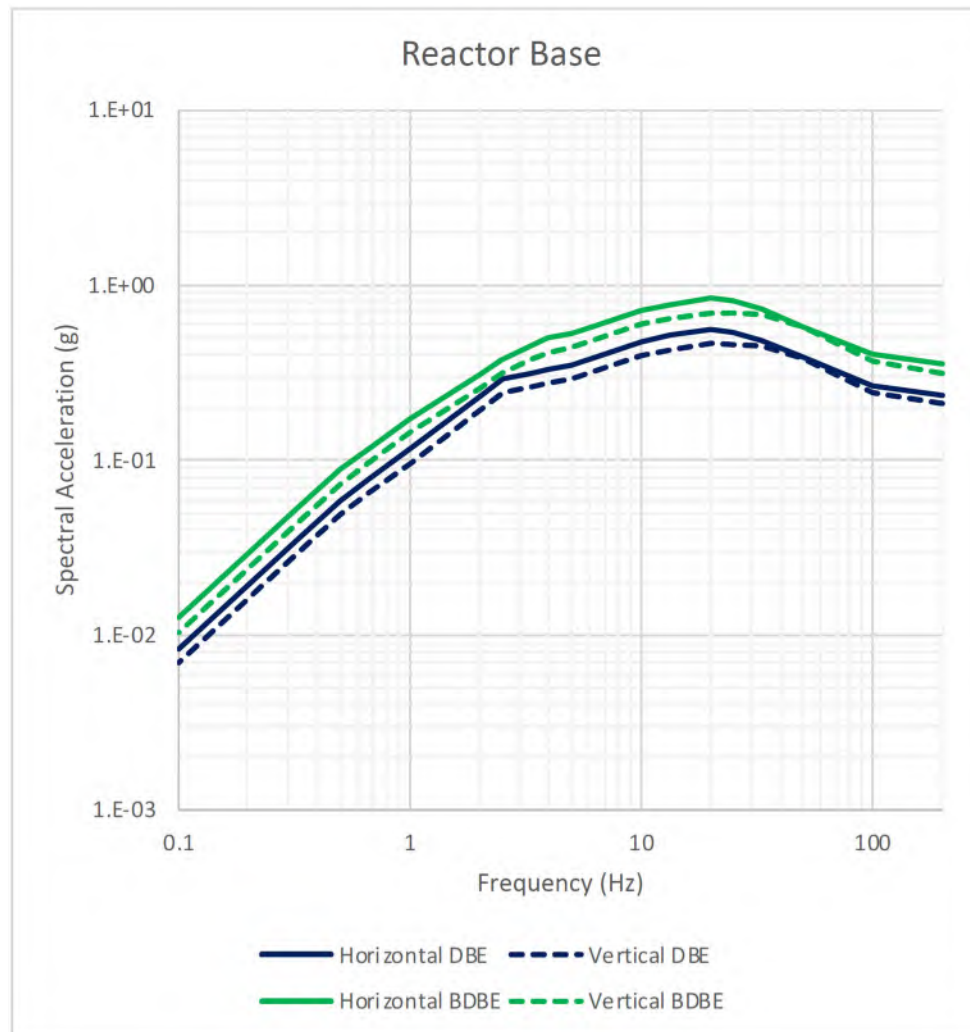
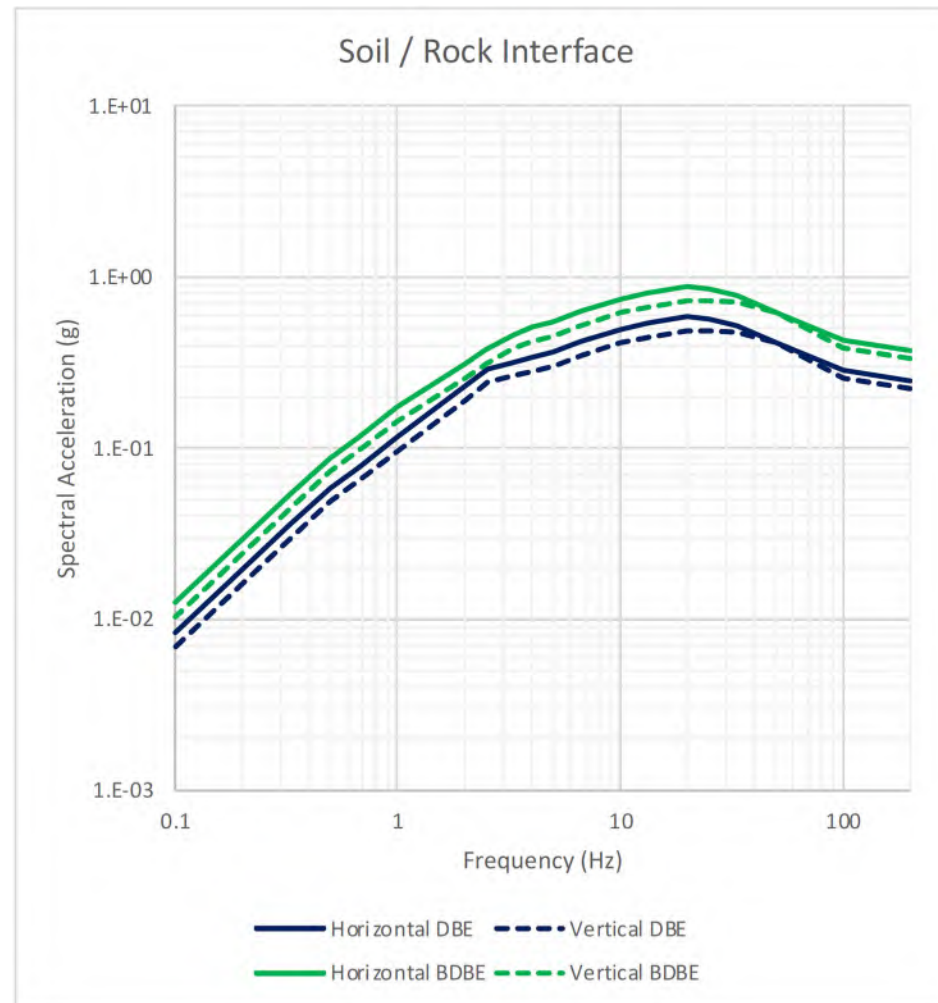
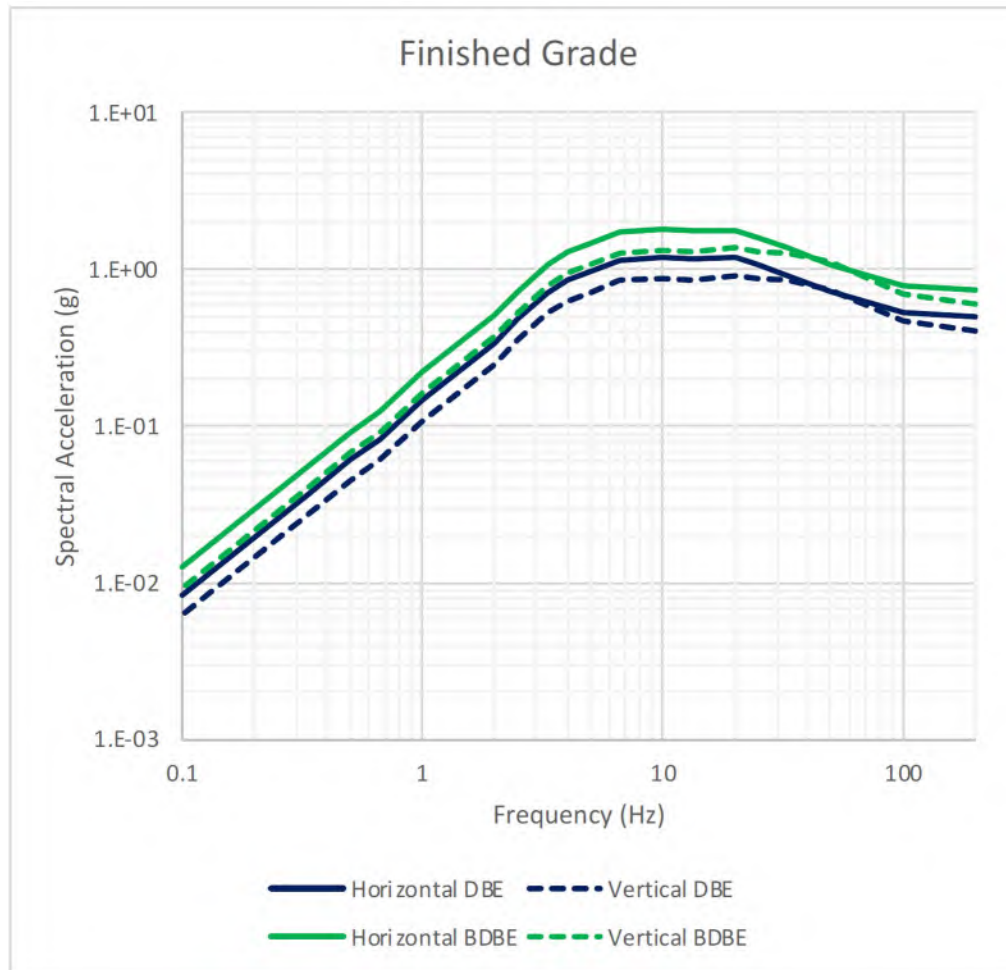


Figure 2.7.5.2.5-1: DBE and BDBE Foundation Input Response Spectra for Reactor Base (Elevation 52.93 m) (Reference 2.7-41)

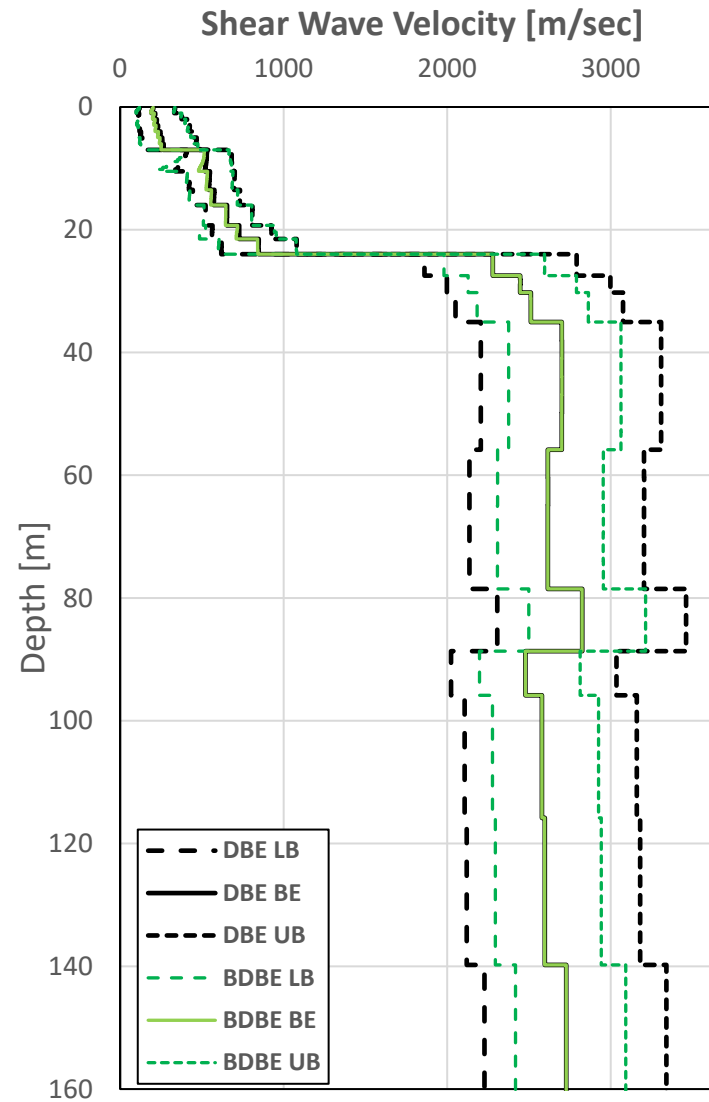


**Figure 2.7.5.2.5-2: DBE and BDBE Performance Based Intermediate Response Spectra for Soil/Rock Interface (Elevation 64 m) (Reference 2.7-41)**

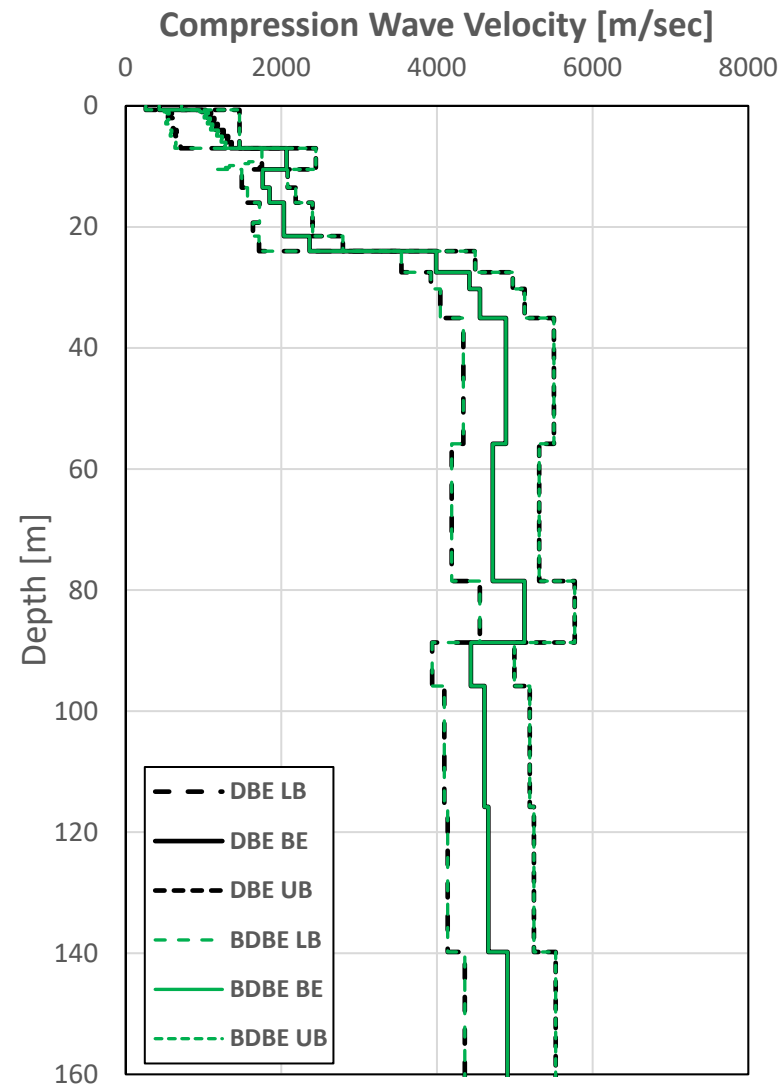




**Figure 2.7.5.2.5-3: DBE and BDBE Performance Based Surface Response Spectra for Finished Grade (Elevation 88 m)  
(Reference 2.7-41)**



**Figure 2.7.5.2.5-4: Subgrade Profiles of DBE and BDBE HCSC Shear Wave Velocities (Reference 2.7-41)**



**Figure 2.7.5.2.5-5: Subgrade Profiles of DBE and BDBE HCSC Compression Wave Velocities (Reference 2.7-41)**

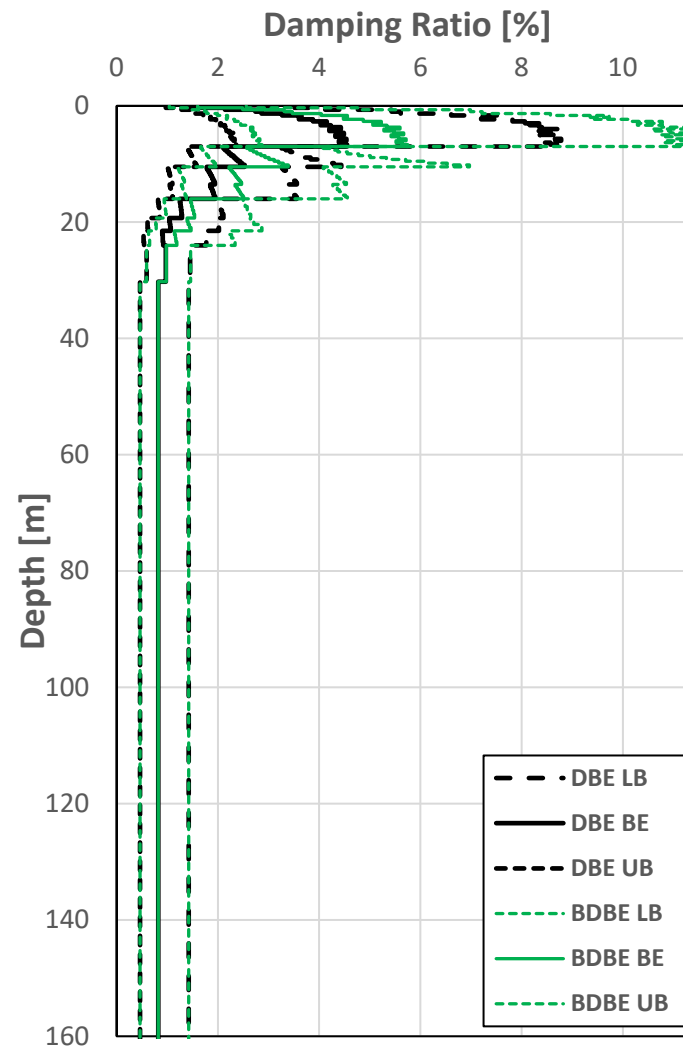


Figure 2.7.5.2.5-6: Subgrade Profiles of DBE and BDBE HCSC Damping Ratios (Reference 2.7-41)

## **2.8 Site Characteristics Impact on Dispersion of Radioactive Material**

The dispersion of radioactive material in water, air, and soil is affected by natural and physical characteristics of the site and the surrounding environment, including meteorology and climate, hydrological and hydrogeological parameters, as well as land cover and use (e.g., vegetation and structures). Population and receptors also influence the potential effects of dispersion of radioactive material. The baseline conditions for these characteristics are established in the:

1. Darlington New Nuclear Project (DNNP) Environmental Impact Statement (EIS), completed in 2009 in NK054-REP-07730-00029 (Reference 2.8-1)
2. Updates to the baseline conditions since the EA was conducted, as discussed in detail in documentation including the 2020 Environmental Risk Assessment (ERA) for the Darlington Nuclear Site, D-REP-07701-00001 (Reference 2.8-2)
3. Yearly Environmental Monitoring Program (EMP) reports, per N-REP-03443-10027 (Reference 2.8-3)
4. DNNP – Site Preparation Licence Renewal Activity Report – Environment, completed in 2020 in NK054-REP-01210-00110 (Reference 2.8-4)
5. Darlington New Nuclear Project Supporting Environment Studies – Environment, completed in 2020, NK054-REP-01210-0001 (Reference 2.8-5)
6. Darlington New Nuclear Project Environmental Impact Statement (EIS) Review Report For Small Modular Reactor BWRX-300, completed in October 2022, per NK054-REP-07730-00055 (Reference 2.8-10)

The 2020 DNNP Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.8-4) concluded the baseline conditions have not changed since the DNNP EA that was conducted in 2009 NK054-REP-07730-00029 (Reference 2.8.1) – a conclusion that is confirmed in the 2022 EIS Review Report NK054-REP-07730-00055 (Reference 2.8-10).

The impact of baseline characteristics of the DNNP site and surrounding environment on dispersion of radioactive material are summarized as follows:

- Impact of meteorology and climate, including Temperature Normals, Precipitation Normals, and Wind Speed and Direction – Subsection 2.8.1
- Impact of hydrology and hydrogeology – Subsection 2.8.2
- Impact of land cover and use – Subsection 2.8.3
- Impact of population, including numbers, locations, ages, and critical groups – Subsection 2.8.4
- Impact of accident scenarios and dispersion models – Subsection 2.8.5
- Impact of biological data – Subsection 2.8.6

Table 2.8-1 lists key characteristics and parameters within the Survey Areas of 10 km and 30 km of the Darlington Nuclear site that encompasses both the Darlington Nuclear Generating Station (DNGS) and DNNP sites.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters**

Characteristic	Value/Description		
2.8.1 Meteorology and Climate			
2.8.1 Climate	Humid with four distinct seasons, uniform precipitation year-round, delayed spring and autumn, moderate temperatures in winter and summer		
2.8.1.1 Temperature Normals	Local Oshawa/Bowmanville Meteorological Stations) Mean Highest	July 4-y monthly average	21.5 °C
	Local (Oshawa/Bowmanville Meteorological Stations) Mean Lowest	January 4-y monthly average	-4.1 °C
	Regional (Toronto Meteorological Station) Mean Highest	July 4-y monthly average	21.5 °C
	Regional (Toronto Meteorological Station) Mean Lowest	January 4-y monthly average	-4.1 °C
	Mean Daily Maximum	August 2016	23.0 °C
	Mean Daily Minimum	January 2019	-6.4 °C
2.8.1.2 Precipitation Normals	Average annual	866 mm (of which <11% snowfall)	
	Total monthly average	From 50.5 mm in February to 98.7 mm in September	
2.8.1.3 Wind Speed and Direction	Predominant (Average wind frequency at 10m height)		ENE (wind from WSW)
	Average Speed		2.4 m/s (Calm winds of <2 m/s were reported 37% of time)
		Direction Wind Blowing From	Darlington Nuclear Wind Frequency (%)
		N	7.22
		NNE	3.09
		NE	3.65
		ENE	8.48
		E	8.25
		ESE	4.60
		SE	3.43
		SSE	2.25
		S	2.33
		SSW	2.35
		SW	6.65
		WSW	9.18
		W	9.98
		WNW	8.34
		NW	9.82
	NNW	10.38	
	Total	100	

**Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters**

Characteristic	Value/Description	
2.8.2 Impact of Hydrology and Hydrogeology		
2.8.2.1 Impact of Hydrology	Lake current	Easterly – near shore Speed in all direction 9 to 18 cm/s
	Lake water temperature	Surface – Freezing to 20 °C Ambient (Winter) – 0.5 °C in January to 7.7°C in November
	Surface Drainage	South of railway – slopes toward Lake Ontario Northeast of railway – slopes toward the east
	Stormwater	Collected in natural channels or swales and constructed outfalls and conveyed to the lake; or ponds
2.8.2.2 Existing Hydrogeological Conditions	Groundwater aquifers	South of railway – north to south Northeast of railway – toward the east Flows are impacted by subsurface structures of BWRX-300 facility.
	Urban areas water supply	Municipal water supply for Lake Ontario
	Rural areas water supply	Surface water intake (lakes) or ground water wells
2.8.3 Impact of Land Cover and Use		
Terrain Type – Water	Lake Ontario – South of the site from the E to the WSW sectors	
Terrain Type – Ploughed land	Within 3 km – Open grassland, farmland, residential homes, parking lots, and industrial land with low-elevation or low-density buildings to the north of the site from the W to the ENE sectors	
Cities	All are farther than 3 km: W and WNW – Oshawa, Whitby, NW – Courtice, and NE – Bowmanville	
Rural Areas	With tall trees, North of the site – NW to NNE, and ENE sectors	
Ecological Features	Meadow (24%), thicket (14%), woodland (5%), and swamp (5%)	
Vegetation communities	Bluff communities	West and east – cover <1% of the Darlington Nuclear site, shrubs with 10% tree cover
	Beach communities	Cover <1% of the Darlington Nuclear site, exposed to the lake with patchy vegetation cover
	Forested areas	Cover about 3% of the Darlington Nuclear site, with 60% tree cover with variable substrate types and conditions
	Cultural communities (resulting from cultural or anthropogenic disturbances)	Cover much of the site, include meadows (24%), thickets (14%), woodlands (5%)

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters**

Characteristic	Value/Description	
	March areas and swamps	3.7% and 5.4%, respectively of the Darlington Nuclear site
Land use in Durham Region and Clarington Municipality	Variety of landscape and lakeshore communities of small rural towns, as well as villages, hamlets, and farm holdings in the northern portion	
	Residential, industrial, and commercial areas	Generally located in Courtice (6.4 km NW of the site), and Bowmanville (4 km NE of the site)
	Agriculture	Predominant land use in Clarington
<b>2.8.4 Impact of Population</b> (Based on Site-specific Survey (2018) and Pathway Analyses (2016))		
Numbers (2016 census)	Within 30 km	<ul style="list-style-type: none"> <li>- Approximately 500,000 within 30 km radius (88% WSW to NNW, 12% E to NE, and 0.0% [Lake Ontario] SW to E of the site)</li> <li>- 90% of population reside in urban areas</li> </ul>
	Within 10 km	Approximately 100,000 residents
	0 to 2.0 km	Only 20 residents
By age (2016 census)	Durham Region	Children (aged under 15) (18%), Young persons (aged 15-29) (19%), Adults (aged 30-64) (49%), Older adults (aged 65+) (14)
By Gender (2018 survey)	Ontario	Largest age group is 20 to 24 for males; 55 to 59 for females
	Durham Region	Largest age group is 50 to 59 for males; 50 to 54 for females
Public Dose Assessment	Critical Groups (site-specific surveys) (NOTE: Annual site-specific survey reports dose for the top three critical groups, as well as specifically for the dairy farm potential critical group)	1. Rural Residents 2. Oshawa/Courtice Residents 3. Bowmanville Residents 4. Local Farms 5. Local Dairy Farms 6. West-East Beach Residents 7. Darlington Provincial Park Campers 8. Sport Fisher 9. Industrial/Commercial Workers
	Site-specific survey (2018) and pathway analyses (2016)	Done about every 5 years Within each critical group, 3 age classes are used – 0-5 years (Infant), 5 to 15 years (child), 16 to 70 years (adult) Group and age classes with highest dose are reported as the site dose for the given year
<b>2.8.5 Impact of Accident Scenarios and Dispersion Models</b>		



**Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters**

Characteristic	Value/Description
	Refer to Chapter 15, Section 15.5 for DBAs and DECAs with and without core melt; as well as events related to irradiated fuel pool and fuel handling
	<b>2.8.6 Impact of Biological Data</b> The baseline terrestrial flora, fauna, and food chain data as well as baseline aquatic biota and food chain data were updated in 2020 in NK054-CORR-00531-10533 (Reference 2.8-9) and did not change the conclusion of the 2009 EIS of NK054-REP-07730-00029 (Reference 2.8.1) as evidenced in the 2022 EIS documented in NK054-REP-07730-00055 (Reference 2.8-10)

## 2.8.1 Impact of Meteorology and Climate

Meteorological characteristics are relevant to the dispersion of material in water, air, and soil as they directly impact the characteristics of the plume, including distance, direction, deposition, and ground concentrations. Relevant meteorological characteristics include temperature, precipitation as well as wind speed and direction.

The Darlington Nuclear site is in Southern Ontario on the north shore of Lake Ontario (refer to Subsection 2.1.1 for additional information). The Darlington Nuclear site displays a humid continental climate with four distinct seasons. In general, Southern Ontario climate is highly modified by the influence of the Great Lakes which results in uniform precipitation amounts year-round, delayed spring and autumn, and moderated temperatures in winter and summer, as described in D-REP-07701-00001 (Reference 2.8-2).

Refer to Section 2.6 for additional DNNP site information relevant to local and regional meteorological characteristics, hazards from meteorological events, and extreme values of meteorological parameters.

### 2.8.1.1 Temperature Normals

The most recent Canadian Climate Normals available span the 1981-2010 period. The meteorological stations at Oshawa and Bowmanville represent the local climate conditions at the Darlington Nuclear site, while the meteorological station at Toronto's Pearson Airport represents the regional conditions. The highest mean temperatures, both regionally and locally, occurred in July, and the lowest mean temperatures occurred in January, as shown in Table 2.8-2. Similar to the local and regional conditions, the highest (21.5 °C) and the lowest (-4.1 °C) 4-year average monthly temperatures at the Darlington Nuclear site occurred in July and January, respectively. The mean daily maximum temperature (23.0 °C) was recorded in August 2016, and the mean daily minimum temperature (-6.4 °C) was recorded in January 2019, as reported in D-REP-07701-00001 (Reference 2.8-2).

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.8-2: Temperature Normals Near the Darlington Nuclear Site (Reference 2.8-2)**

Month	Daily Mean (°C)				Mean Daily Maximum (°C)				Mean Daily Minimum (°C)			
	Regional Study Area	Local Study Area		Site Study Area	Regional Study Area	Local Study Area		Site Study Area	Regional Study Area	Local Study Area		Site Study Area
	TOR <sup>1</sup>	OSH <sup>2</sup>	BOW <sup>3</sup>	DN <sup>4</sup>	TOR <sup>1</sup>	OSH <sup>2</sup>	BOW <sup>3</sup>	DN <sup>4</sup>	TOR <sup>1</sup>	OSH <sup>2</sup>	BOW <sup>3</sup>	DN <sup>4</sup>
January	-5.5	-4.8	-5.6	-4.1	-1.5	-1.1	-1.4	-1.5	-9.4	-8.5	-9.9	-6.4
February	-4.5	-3.6	-4.4	-2	-0.4	0.1	0	-0.5	-8.7	-7.3	-8.8	-4.1
March	0.1	0.4	-0.2	-0.1	4.6	4.2	4.3	2	-4.5	-3.5	-4.6	-1.1
April	7.1	6.6	6.4	5.4	12.2	10.8	11.3	8.3	1.9	2.5	1.5	3.2
May	13.1	12.3	12.4	12.6	18.8	16.9	18	13.8	7.4	7.7	6.8	11.6
June	18.6	17.6	17.5	17.8	24.2	22.3	23.1	18.3	13	12.9	11.8	17.4
July	21.5	20.6	20	21.5	27.1	25.1	25.8	22.1	15.8	15.9	14.3	20.7
August	20.6	20	19.2	21.3	26	24.3	24.8	23	15.1	15.6	13.5	19.5
September	16.2	15.9	15	18	21.6	20.2	20.4	18.8	10.8	11.7	9.5	16.8
October	9.5	9.5	8.7	11.2	14.3	13.3	13.7	13.2	4.6	5.6	3.6	9.1
November	3.7	4.2	3.4	3.3	7.6	7.4	7.2	6.3	-0.2	1	-0.4	1.1
December	-2.2	-1.2	-2.2	-1.7	1.4	2.1	1.6	-0.1	-5.8	-4.4	-6	-5.2
Year	8.2	8.1	7.5	8.6	13	12.1	12.4	10.3	3.3	4.1	2.6	6.9

1. Toronto Lester B. Pearson International Airport, 1981-2010 Climate Normals

2. Oshawa Water Pollution Control Plant (WPCP), 1981-2010 Climate Normals

3. Bowmanville Mostert Station, 1981-2010 Climate Normals

4. Darlington Nuclear, 2016-2019 (2017 data from Darlington Nuclear site on-site meteorological tower, while 2016, 2018, 2019 data from Pickering Nuclear on-site meteorological tower).

### 2.8.1.2 Precipitation Normals

The Bowmanville climate station is the closest to the Darlington Nuclear site. The 1981-2010 climate normal precipitation data, listed in Table 2.8-3, from the Bowmanville Mostert Station are used to characterize precipitation patterns for the Darlington Nuclear site. During this period the Bowmanville station reported an average annual precipitation of approximately 866 mm; with snowfall representing less than 11% of the total precipitation measured. Total monthly precipitation averages range from approximately 50.5 mm in February to approximately 98.7 mm in September, per D-REP-07701-00001 (Reference 2.8-2).

**Table 2.8-3: Precipitation at Bowmanville Mostert Station (1981-2010)**

Month	Monthly Averages			Daily Extremes		
	Precipitation (mm)	Rain (mm)	Snow (cm)	Precipitation (mm)	Rain (mm)	Snow (cm)
January	63.1	32.2	31	46.2	46.2	29
February	50.5	32.8	17.7	42.2	42.2	19.4
March	55	41	14.1	47.6	47.6	20.8
April	70.6	68	2.6	43.4	43.4	10.2
May	75.9	75.9	0	36.4	36.4	0
June	83.8	83.8	0	50.6	50.6	0
July	63.2	63.2	0	51.1	51.1	0
August	78.1	78.1	0	81.2	81.2	0
September	98.7	98.7	0	84	84	0
October	70.8	70.6	0.1	48.6	48.6	12.2
November	88.6	83.1	5.6	71.4	71.4	15.5
December	68.1	46.1	22	41.1	41.1	24
Annual Total	866.4	773.5	93.1	-	-	-

### 2.8.1.3 Wind Speed and Direction

As discussed in the annual EMP report N-REP-03443-10027 (Reference 2.8-3), the wind speed, direction, and frequency are measured continuously at meteorological towers at the Darlington Nuclear site. As shown in Table 2.8-4 for the year 2021, the landward sector at the Darlington Nuclear site the wind predominantly blew toward was the ENE sector (wind from WSW), based on the average annual wind frequencies at a 10 m height. Over all sectors, the wind predominantly blew from the north and west sectors. The dominant wind direction was NNW (10.38% of the time), followed by W (9.98% of the time) and NW (9.82% of the time).

**Table 2.8-4: Darlington Nuclear – 2021 Annual Average Wind Frequency by Direction (at 10 m height)**

<b>Direction Wind Blowing From</b>	<b>Darlington Nuclear Wind Frequency (%)</b>
N	7.22
NNE	3.09
NE	3.65
ENE	8.48
E	8.25
ESE	4.60
SE	3.43
SSE	2.25
S	2.33
SSW	2.35
SW	6.65
WSW	<b>9.18</b>
W	9.98
WNW	8.34
NW	9.82
NNW	10.38
Total	100

Notes:

- (1) Shaded fields indicate landward wind sectors.
- (2) Bolded values indicate landward wind sectors with the highest wind frequency.

As reported in the 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.8-2), wind speeds were measured from 2013-2019 at the Darlington Nuclear on-site meteorological towers at a height of 10 m. The average wind speed was approximately 2.4 m/s. Calm winds of less than 2 m/s were reported approximately 37% of the time. The prevailing winds for these years were measured to be from the north-west sector – the north direction (9.6% of the time) followed by the west direction (8.9% of the time). The wind rose for the 2013-2019 data is provided in Figure 2-8 of D-REP-07701-00001 (Reference 2.8-2).

## **2.8.2 Impact of Hydrology and Hydrogeology**

Hydrological and hydrogeological characteristics are relevant to the dispersion of material in water. These characteristics influence the flow and concentration of radioactive and conventional contaminants, as well as impact the populations that are affected. Relevant characteristics include aquifer type, groundwater flow, stormwater runoff, municipal water supply sources, lake currents and temperature, and major lake water intake and discharge structures.

Refer to Section 2.5 for further information on the implication of hydrological and hydrogeological conditions, including abnormal phenomena at the DNNP site on the design and safe operation of the BWRX-300 facility.

### **2.8.2.1 Impact of Hydrology**

There is very little current net flow along the northern shore of Lake Ontario. However, the current in the nearshore region is overall easterly and is influenced by brief patterns of strong winds exerting stress at the water surface. Lake current speeds for all directions for the 2012-2016 period typically ranged from about 9 to 18 cm/s and were typically slower during spring and early summer, (May through June) than during late summer, fall and winter (August through April), as described in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

Lake-wide surface temperatures typically range from freezing in the winter to approximately 20 °C in the summer. Ice formation in the winter is typically limited to the nearshore areas at the eastern end of the lake within the Kingston Basin. Average ambient water temperatures in the winter have varied from 0.5 °C in January to 7.7 °C in November. The water temperatures recorded from December 2011 to March 2012 and from December 2011 to April 2012 in the Darlington Nuclear study area had an average temperature of 3.8 °C and 4.4 °C, respectively, per the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

The intake pumphouse/forebay of the BWRX-300 facility provides the transition of water flowing from the intake tunnel up to the Circulating Water System pumps (refer to Subsection 2.5.2) via an onshore vertical shaft. The intake offshore tunnel transitions into a porous veneer intake. Similarly, the submerged discharge tunnel connects to a discharge shaft that is located near the shoreline bluff, to convey returned heated water to the diffusers. Refer to Chapter 9B, Subsection 9B.3.5, for design information on the BWRX-300 pumphouse/forebay, intake and discharge shafts and tunnels, lakebed intake structure and discharge diffusers.

The surface drainage at the Darlington Nuclear site is divided by the Canadian National Railway line which runs east to west across the site (refer to Section 2.1, and Figure 2.1.1.2). The area south of the railway tracks generally slopes toward Lake Ontario while the area north of the railway tracks and east of Holt Road slopes toward the east. In the developed parts of Darlington Nuclear site including the DNGS areas, stormwater is collected in natural channels/swales and constructed outfalls and conveyed to Lake Ontario. Currently, a stormwater pond is located to the south of the Engineering Support Services Building and another pond is associated with the Darlington Waste Management Facility (DWMF). Another stormwater pond is located north of the lagoons which collect runoff from adjacent parking lots and from the railroad tracks (refer to the 2020 ERA D-REP-07701-00001 (Reference 2.8-2)). These features could change as the DNNP site is further developed, and the BWRX-300 design progresses.

To support the Site Preparation Licence renewal application in 2020, OPG obtained hydrological data, surface water data, and sediment quality data in the site, as well as in the local, and regional study areas, as provided in the 2009 DNNP EIS NK054-REP-07730-00029 (Reference 2.8-1).

The 2022 EIS in NK054-REP-07730-00055 (Reference 2.8-10) reports that the BWRX-300 deployment will have no residual adverse effects on site drainage and identified minor changes

in DNNP flows and the number of days per year that an area of land is wet can be mitigated using best industry practices.

#### **2.8.2.2 Existing Hydrogeological Conditions**

The information on existing groundwater conditions discussed in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2) and the 2009 DNNP Supporting Environment Studies NK054-REP-01210-0001 (Reference 2.8-5) is detailed in Subsection 2.5.5.

Inside the protected area at DNGS, groundwater flow is further influenced by anthropogenic subsurface features such as foundations, drain systems and sumps, and the vacuum building.

For the protected area at the DNNP, the Power Block footprint is smaller than the DNGS footprint. Also, the Reactor Building (RB) is embedded in the soil and extends to bedrock, impacting connection between groundwater flows at the north and south of the structure, per the 2020 ERA D-REP-07701-00001 (Reference 2.8-2). Such anthropogenic DNNP structures would influence the hydrostratigraphic layers and the neighboring groundwater flows. (Refer to Chapter 1, Subsection 1.5.2, and Table 1.5.2 for dimensions of the RB and other buildings in the Power Block).

Recharge of precipitation is expected to be low at the Darlington Nuclear site in areas where till is encountered at surface. Within these areas most precipitation runs off to surface water ditches or yard drainage features, as described in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2). (Refer to Subsection 2.5.3 for additional information on potential sources of flooding).

Since the Site Preparation Licence renewal application in 2020 included in NK054-CORR-00531-10533 (Reference 2.8-9), OPG examined groundwater flow characteristics at the Darlington Nuclear site as part of annual groundwater monitoring (refer to Subsection 2.5.5.3). Furthermore, additional geotechnical investigations are completed for the DNNP's onshore Power Block area, with the results documented in the 2022 NK054-REP-10180-00001 DNNP Geotechnical and Seismic Hazard Investigation Plan – Phase 1 (Reference 2.8-11).

Groundwater on the Darlington Nuclear site is not used as drinking water and is not considered to be potable.

Annual groundwater quality monitoring (described in Subsection 2.5.5.3) is carried out across the site study area. Recent monitoring results, such as the levels of tritium, Volatile Organic Components, Benzene, Toluene, Ethylbenzene and Xylene, Petroleum Hydrocarbons, sodium, chloride, and metals in groundwater, are used to establish the groundwater quality baseline. Based on the annual groundwater monitoring results for the period of 2019 to 2021, groundwater quality remains consistent with that documented in the licence to prepare site application, per the 2020 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.8-4). The tritium concentrations at the sampled perimeter groundwater locations remained low in 2021. This is aligned with a trend observed indicating the tritium levels over time have remained nearly steady or decreased, which indicates stable or improved environmental performance. The groundwater quality results were compared to the Ministry of Environment, Conservation and Parks' Provincial Water Quality Objectives, based on the assumption that groundwater pumped during construction or in the long term will be discharged to the natural environment. Some groundwater samples exhibited elevated concentrations of total metals, dissolved metals, phenols, and toluene above the selected Provincial Water Quality Objectives. Several samples exhibited pH outside the acceptable Provincial Water Quality Objectives range of 6.5 to 8.5. However, given that the water is not used for drinking and is not considered potable, the conclusions of the original Site Evaluation, reported in the 2020 renewal of licence to prepare site application NK054-REP-01210-00110 (Reference 2.8-4), are valid.

Urban areas such as Bowmanville to the east and Courtice to the west of the Darlington Nuclear site rely on municipal water supply from a Lake-Ontario-based source. The more rural areas of Durham are supplied by individual water supply systems from either surface water intakes (lakes) or ground water wells. There are rural and farm residents in rural areas in all landward wind sectors around the site at distances of about 2 km to 5 km. Residents in these areas obtain at least a portion of their water supply from wells, and use it for drinking, bathing, and irrigation. However, there are no potable groundwater supply wells within or downgradient of potential source areas on-site. As water on the Darlington Nuclear site is not used for human consumption, the only on-site pathway for human exposure to groundwater would be from ingestion of water from Lake Ontario after dilution of the groundwater in the lake. Off-site drinking water wells are influenced by atmospheric tritium, but this makes a negligible contribution to dose. Concentrations of potential chemical stressors in off-site drinking water wells are not influenced by the Darlington Nuclear site, refer to the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

### **2.8.3 Impact of Land Cover and Use**

Land cover and use characteristics are relevant to the dispersion of material in water, air and soil as these characteristics define the terrain cover and impact deposition. Relevant characteristics include terrain type, vegetation type, vegetation height, building height, and locations.

The terrain cover surrounding the Darlington Nuclear site is broadly characterized for air dispersion modelling (refer to Subsection 2.8-5) in the Derived Release Limits and Environmental Action Levels for DNGS NK38-REP-03482-10001 (Reference 2.8-6). The major terrain types are as follows:

- Water: Lake Ontario to the south of the site from the E to the WSW sectors
- Ploughed Land: At the site boundary to a distance of 3 km, open grassland, farmland, residential homes, parking lots, and industrial land with low-elevation or low-density buildings to the north of the site from the W to the ENE sectors

At distances further than 3 km from the site boundary, inspection of aerial photographs shows cities with larger buildings, including Oshawa and Whitby to the W and WNW of the site, and Bowmanville to the NE of the site. Rural areas with tall trees, including Ganaraska Forest, are located north of the site from the NW to the NNE sectors and ENE sectors.

The dominant ecological feature of the Darlington Nuclear site is meadow (24%), followed by thicket (14%), woodland (5%), and swamp (5%). In general, the Darlington Nuclear site has four main areas, per NK054-REP-01210-0001 (Reference 2.8-5):

1. In the northwest there are sports fields, a large settling pond (Coot's Pond), and Bobolink Hill comprised of cultural meadow and cultural thicket
2. In the northeast there are agricultural fields, cultural thicket, and deciduous forest as well as three constructed wetland ponds (Treefrog, Dragonfly and Polliwog ponds)
3. In the southeast there are mostly cultural meadows
4. In the south centre and southeast is the DNGS

There are various terrain types and vegetation communities on or immediately surrounding the Darlington Nuclear site, including bluffs, beach, forest, cultural woodland, cultural meadow, cultural thickets, marshland, swamp, and urban areas. The dominant vegetation cover surrounding the Darlington Nuclear site relates to agricultural use, including row crops and pastureland, as detailed in D-REP-07701-00001 (Reference 2.8-2).

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Bluff communities are present west and east of the Darlington Nuclear site and cover a very small portion (<1%) of the Darlington Nuclear site. Bluff communities are characterized by variable vegetation cover that can range from patchy and barren to herbaceous cover. Generally, bluffs have no more than 10% tree cover because of erosion which results in steep, sometimes near vertical faces that are more than 2 meters in height. The bluff community on the west side of the Darlington Nuclear site is dominated by shrubs, mostly willows with Red-Osier Dogwood and Nannyberry. The bluff community on the east side of the Darlington Nuclear site is characterized by open or sparsely vegetated land due to ongoing erosional disturbance. The most abundant vegetation on these bluffs is Colt's Foot, refer to D-REP-07701-00001 (Reference 2.8-2).

The beach community covers a very small fraction (<1%) of the Darlington Nuclear site and much of the area is relatively exposed to the lake. The beach community is characterized by patchy vegetation cover that varies from sparse cover to areas with treed cover equal to or less than 60%, as described in D-REP-07701-00001 (Reference 2.8-2).

Forested areas cover about 0.16 km<sup>2</sup> (about 3%) at the Darlington Nuclear site. The forest community is characterized by a high level of tree cover (more than 60%) as well as variable substrate types and conditions and is classified as a coniferous, deciduous, or mixed forest type, as detailed in D-REP-07701-00001 (Reference 2.8-2).

Much of the Darlington Nuclear site vegetation communities are characterized as cultural communities such as cultural meadows, thickets, and woodlands (including plantations) that generally resulted from or are maintained by cultural or anthropogenic disturbances. Cultural woodlands, meadows, and thickets arise following anthropogenic disturbance. Cultural woodlands cover approximately 5% of the Darlington Nuclear site. They are characterized by a relatively open canopy (less than 60% cover). Cultural meadows cover approximately 24% of the Darlington Nuclear site. There are many types of cultural thickets that cover approximately 14% of the Darlington Nuclear site. They are formed during early successional stages following anthropogenic disturbance. Shrubs generally comprise the bulk of the vegetation cover and include a high proportion of non-native species, refer to D-REP-07701-00001 (Reference 2.8-2) for additional information.

Marsh areas cover over approximately 0.2 km<sup>2</sup> on the Darlington Nuclear site, or 3.7% of the total area. Swamp areas are the most dominant of the Wetland Community Classes at the Darlington Nuclear site, covering approximately 0.25 km<sup>2</sup>, or 5.4% of the total Darlington Nuclear site. Swamps are characterized by the presence of wetland trees and shrubs and a low proportion of tree and shrub cover, as reported in D-REP-07701-00001 (Reference 2.8-2).

Durham Region is characterized by a variety of landscapes and communities including major lakeshore urban communities in the southern portion, and small rural towns, villages, hamlets and farm holdings in the northern portion of the region. Urban land uses are generally parallel the shoreline of Lake Ontario in the communities of Pickering, Ajax, Whitby, Oshawa and Clarington, while rural land uses are found in the communities of Brock, Scugog and Uxbridge in the northern portion of the region, all are described in D-REP-07701-00001 (Reference 2.8-2).

Urban land uses in the Municipality of Clarington, including residential, commercial, and industrial, are generally located in Courtice, located approximately 6.4 km northwest of the Darlington Nuclear site, and Bowmanville, located approximately 4 km northeast of the site. Agriculture is a predominant land use in the Municipality of Clarington and is less predominant in the City of Oshawa west of the site, per D-REP-07701-00001 (Reference 2.8-2). (Refer to Subsection 2.1.1 for recent and forecast land use data for the Municipality of Clarington and the City of Oshawa.)



#### **2.8.4 Impact of Population**

Population characteristics are relevant to the determination of the potential effects of the dispersion of material in water, air, and soil as the dispersion of radioactive and conventional contaminants affects the population surrounding the Darling Nuclear site. Relevant characteristics include population numbers, locations, ages, and critical groups.

The census data for the region used in the most recent Review of the Darlington Nuclear Site-Specific Survey, reported in NK38-REP-03443-10004 (Reference 2.8-7), are for 2016.

A population of approximately 500,000 resides within a 30 km radius of the Darlington Nuclear site, based on 2016 census data shown in Table 2.8-5. The bulk of this population (approximately 88% or 478,634 individuals) resides west of the Darlington Nuclear site, in the west-south-west to north-north-west sectors, while approximately 12% (64,575 individuals) reside east of the Darlington Nuclear site in the north to east north-east sectors. Areas south and east of the Darlington Nuclear site (south-west to east) are occupied by Lake Ontario. Only 20 residents reside within a 0 to 2 km radius of the centre of Darlington Nuclear site and approximately 99,953 individuals reside within 10 km of the Darlington Nuclear site, as documented in D-REP-07701-00001 (Reference 2.8-2).

The majority of residents in the Durham Region live in urban areas. Over 90% of the population in Pickering, Ajax, Oshawa, and Whitby resides in urban areas, whereas, the townships of Brock, Scugog and Uxbridge represent the greatest percentage of the rural population in Durham. Urban/rural population trends for Durham indicate this trend will continue into 2031, per D-REP-07701-00001 (Reference 2.8-2).

Children under the age of 15 comprised 18.0% of Durham's population in 2016, while young persons (aged 15-29), adults (aged 30-64) and older adults (aged 65+) comprised 19.2%, 49.4% and 14.4%, respectively. Ontario Population Estimates for 2018 indicate the 20 to 24 age group is the largest age group for males and 55 to 59 for females in Ontario, while in Durham Region the largest age group was 50 to 59 for males and 50 to 54 for females, refer to D-REP-07701-00001 (Reference 2.8-2).

In public dose assessments, "critical groups" are used to estimate the mean realistic impacts of emissions on the most affected individuals. The site-specific surveys identify the potential critical groups for Darlington Nuclear site. Approximately every five years the site-specific surveys and pathway analyses are reviewed to ensure the public dose accurately represents the public living near Darlington Nuclear site. Site-specific surveys were most recently reviewed in 2018 and pathway analyses were last updated in 2016. The EMP design reviews were conducted in 2018, and minor changes are implemented in 2019 which primarily affect which potential critical groups are used for reporting purposes, as documented in N-REP-03443-10027 (Reference 2.8-3).

An individual with the average characteristics of the critical group is known as the "Representative Person" as described in CSA N288.1-14 (Reference 2.8-8). Dose estimates are calculated for a number of potential critical groups for Darlington Nuclear site, and for three age classes within each potential critical group. The three age classes are 0-5 years (infant), 6-15 years (child), and 16-70 years (adult). The dose estimates to these three age groups are sufficient to characterize doses to the public. For practical implementation in dose calculations, the dose coefficients, and characteristics for a one-year-old infant, a 10-year-old child, and an adult are used to represent the three age classes. The group and age class with the highest dose is reported as the site public dose for the given in year, as described in N-REP-03443-10027 (Reference 2.8-3). (Refer to Subsection 2.9.1.2 for information on radiological dose to the public).

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Nine potential critical groups are identified for the Darlington Nuclear site. The list of potential critical groups around Darlington Nuclear site includes the following, per NK38-REP-03443-10004 (Reference 2.8-7):

1. Rural Residents
2. Oshawa/Courtice Residents
3. Bowmanville Residents
4. Local Farms
5. Local Dairy Farms
6. West-East Beach Residents
7. Darlington Provincial Park Campers
8. Sport Fisher
9. Industrial/Commercial Workers

The annual public dose is calculated for specific three potential critical groups only, which have yielded the highest dose estimates in recent years. These are the Farms, the West/East Beach Residents, and the Rural Residents, as described in N-REP-03443-10027 (Reference 2.8-3). Additionally, the annual public dose is also calculated for the local dairy farm potential critical group as the dairy farm group is exposed to the most media types and pathways.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.8-5: Population Distribution Surrounding Darlington Nuclear Site Based on 2016 Census Data (Reference 2.8-2)**

Direction	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0-2 km	20	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	20
2-4 km	10	3,516	0	0	0	0	0	0	0	0	0	0	0	10	69	50	3,655
4-6 km	1,612	6,803	5,037	315	0	0	0	0	0	0	0	0	0	1,611	1,646	589	17,613
6-8 km	569	14,691	5,809	314	0	0	0	0	0	0	0	0	5	13,936	10,172	247	45,743
8-10 km	751	1,507	196	1,217	0	0	0	0	0	0	0	0	7,389	15,749	5,729	384	32,922
10-12 km	897	221	462	5,004	0	0	0	0	0	0	0	0	15,568	29,781	7,768	251	59,952
12-14 km	390	129	398	3,375	0	0	0	0	0	0	0	0	7,115	27,662	15,599	412	55,080
14-16 km	436	734	943	875	0	0	0	0	0	0	0	0	9,013	21,052	7,294	214	40,561
16-22 km	850	873	691	1,287	0	0	0	0	0	0	0	732	50,773	60,986	4,655	1,394	122,241
22-30 km	1,224	1,562	981	876	0	0	0	0	0	0	0	7,998	141,667	6,853	2,705	1,556	165,422
Total	6,759	30,036	14,517	13,263	0	0	0	0	0	0	0	8,730	231,530	177,640	55,637	5,097	543,209

### **2.8.5 Impact of Accident Scenarios and Dispersion Models**

Accident scenarios and associated dispersion models are described in Chapter 15, Section 15.5, for Design Basis Accidents (DBAs), Design Extension Conditions (DECs) with and without core melt, as well as for irradiated fuel pool and fuel handling events for BWRX-300 site-specific application.

### **2.8.6 Impact of Biological Data**

The biological characteristics of the site were documented in the 2009 DNNP EIS, NK054-REP-07730-00029 (Reference 2.8-1), to support the original application of the Site Preparation Licence. The report includes both baseline of terrestrial flora, fauna and food chain data, as well as baseline aquatic biota and habitat, and food chain data. The biological characterization underwent a baseline update for the 2020 Site Preparation Licence renewal, which is provided in NK054-CORR-00531-10533 (Reference 2.8-9). The 2020 updated baseline conditions will not change the conclusion with respect to residual adverse effects of the on the environment nor the conclusions of the original Site Evaluation. The same conclusion is confirmed the recent 2022 EIS documented in NK054-REP-07730-00055 (Reference 2.8-10).

### **2.8.7 References**

- 2.8-1 NK054-REP-07730-00029 R000, 2009, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.8-2 D-REP-07701-00001 R001, 2020 "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.8-3 N-REP-03443-10027 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.8-4 NK054-REP-01210-00110 R001, 2020, "DNNP – Site Preparation Licence Renewal Activity Report – Environment," Ontario Power Generation.
- 2.8-5 NK054-REP-01210-0001 R000, 2020, "Darlington New Nuclear Project Supporting Environment Studies – Environment," Ontario Power Generation.
- 2.8-6 NK38-REP-03482-10001 R002, "Derived Release Limits and Environmental Action Levels for Darlington Nuclear Generating Station," Ontario Power Generation.
- 2.8-7 NK38-REP-03443-10004 R001, 2021, "Review of the Darlington Nuclear Site-Specific Survey," Ontario Power Generation.
- 2.8-8 CSA N288.1-14, "Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities," CSA Group.
- 2.8-9 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation License (PRSL)," Ontario Power Generation.
- 2.8-10 NK054-REP-07730-00055 R000, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.8-11 NK054-REP-10180-00001 R000, (GOLDER 2022), "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation.

## 2.9 Radiological Conditions Due to External Sources

Section 2.9 details information on:

- Radiological Conditions in the Environment – Subsection 2.9.1, including
  - Radiological Baseline Conditions – Subsection 2.9.1.1
  - Radiological Dose to Public Due to Activities on DNGS Site – Subsection 2.9.1.2
- Radiation Monitoring Systems – Subsection 2.9.2, including
  - Environmental Monitoring Program – Subsection 2.9.2.1
  - TLD Monitoring – Subsection 2.9.2.2
  - Gamma Monitoring – Subsection 2.9.2.3
  - Effluent Monitoring – Subsection 2.9.2.4

Table 2.9-1 lists key characteristics and parameters for the radiological conditions due to sources external to the DNNP site.

**Table 2.9-1: DNNP Site Radiological Conditions in 2021**

Characteristic	Value/Description		
2.9.1 Radiological Conditions in the Environment			
Sources of Baseline radiation and Radioactivity	<ul style="list-style-type: none"><li>Natural background</li><li>Nuclear testing, nuclear facilities</li><li>DNGS, Tritium Removal Facility, DWMF</li></ul>		
Radiological Emissions	Small fraction of the Derived Release Limit (DRL) <ul style="list-style-type: none"><li>2016 to 2019 &lt;0.01 – 0.41% of the DRLs</li><li>In 2021 &lt;0.01 – 0.53% of the DRLs</li></ul>		
2.9.1.1 Radiological Baseline Conditions			
NOTE: The unit Bq/kg-C means becquerels per each kilogram of Carbon			
Air Samples – Concentrations	tritium	Range: 0.2 to 1.8 Bq/m <sup>3</sup>	Average: 0.87 Bq/m <sup>3</sup>
	C-14	Range: 206 to 248 Bq/kg-C	Average: 230 Bq/kg-C
	Ar-41, Xe-133, Xe-135, and Ir-192		Estimated to be below detection
Terrestrial Samples – Concentration	Average tritium	In fruits	17.8 Bq/L
		In vegetables	17.5 Bq/L
		In milk	4.3 Bq/L
		In animal feed	8.6 Bq/L
	Average C-14	In fruits	230 Bq/kg-C
In vegetables		248 Bq/kg-C	
In milk		229 Bq/kg-C	
In animal feed		236 Bq/kg-C	
Soil Sampling in 2017 (every 5 years)	<ul style="list-style-type: none"><li>Cs-137, background values (from 1.7 to 9.0 Bq/kg) are present as results of historic weapon testing and around DNGS (5.1 to 7.2 Bq/kg)</li></ul>		

**Table 2.9-1: DNNP Site Radiological Conditions in 2021**

Characteristic	Value/Description	
		<ul style="list-style-type: none"> <li>Co-60 and Cs-134, due to emission from DNGS and other nuclear sites, neither detected.</li> </ul>
Aquatic Samples – Concentration	tritium	<ul style="list-style-type: none"> <li>All nearby water Supply plants – Average is below provincial standard of 7,000 Bq/L</li> <li>Bowmanville, Newcastle, and Oshawa water supply plants, range from 4.6 to 6.6 Bq/L</li> <li>Well Water – Average 12.0 Bq/L</li> <li>Lake Water – Average 9.6 Bq/L</li> <li>Fish – Average &lt;3.4 Bq/L</li> </ul>
	C-14	<ul style="list-style-type: none"> <li>Fish – Average 243 Bq/kg-C</li> </ul>
	C-137	<ul style="list-style-type: none"> <li>Fish – Average 0.2 Bq/kg</li> <li>Sand Beach – (&lt; 0.1) to 0.2 Bq/kg</li> </ul>
	Co-60 and Cs-124	<ul style="list-style-type: none"> <li>Fish – Not detected</li> <li>Sand Beach – Not detected</li> </ul>
	Gross beta activities	All nearby water Supply plants – Average 1 Bq/L, which is below Health Canada Guideline for drinking water
<p><b>NOTES:</b></p> <ol style="list-style-type: none"> <li>In 2021 ground water monitoring program, tritium concentrations at the sampled Darlington Nuclear site perimeter groundwater locations remained low.</li> <li>In general, tritium trends over time show levels have remained nearly steady or decreased, indicating stable or improved environmental performance</li> <li>Where unexpected tritium concentrations are identified, investigations are completed to determine the root cause and to implement corrective measures.</li> <li>Ongoing results confirm that tritium in groundwater is mainly localized within the station protected area and the site perimeter tritium concentrations remain low</li> </ol>		
<b>2.9.1.2 Radiological Dose to the Public</b>		
<p>Public dose for the Darlington Nuclear site was 0.6 µSv/year (represented by the adult farm resident critical group); which is</p> <ul style="list-style-type: none"> <li>&lt;0.1% of the regulatory limit of 1,000 µSv/year for a member of the public</li> <li>&lt;0.1% of the background radiation around Darlington Nuclear site</li> </ul>		
<b>2.9.2 Radiation Monitoring Systems</b>		
<b>2.9.2.1 Environmental Monitoring Program</b>		
<b>2.9.2.1.1 Atmospheric Sampling</b>	tritium	Active samplers at six site boundary locations. Samples are collected and analysed monthly
	C-14	Monitored at four boundary locations and analysed each quarter
	Noble gases	8 detectors that monitor gamma radiation dose rate continuously

**Table 2.9-1: DNNP Site Radiological Conditions in 2021**

Characteristic	Value/Description	
2.9.2.1.2 <i>Aquatic Sampling</i>	Drinking water	Samples taken every 8-12-hour shift. Weakly composites are analysed weekly for tritium and monthly for gross beta activates
	Well water	Collected from four wells and analysed monthly for tritium
	Lake water	Sampled from two beaches and analysed monthly for tritium
	Fish	At DNGS – Muscle-tissue eight replicated target fish species are collected for tritium, C-14, Co-60, Cs-134, Cs-137, and Potassium-40 (K-40) measurements
	Sand	Samples collected from three beaches and analysed annually using gamma spectrometry to detect Cs-137
	Groundwater	81 monitoring locations are sampled each year for tritium.
2.9.2.1.3 <i>Terrestrial Sampling (tested for tritium and C-14)</i>	Fruits and Vegetables	Sampled three times from each of five locations representing the growing season
	Milk	Samples collected monthly from three dairy farms around the site
	Animal feed	Samples collected form four dairy farms with two replicates per visit. Dry feed and wet feed are collected separately
	Eggs	Sampled quarterly with three samples replicated per visit. Poultry samples collected annually with eight samples replicated per visit
<b>2.9.2.2 Thermoluminescent Dosimeter (TLD) Monitoring</b>		
Located around the site and off-site. TLD cards are analysed annually when they are changed. They are located around the DWMF fence line		
<b>2.9.2.3 Gamma Monitoring System</b>		
The automated fixed monitors provide real-time gamma dose rate measurements		
<b>2.9.2.4 Effluent Monitoring Program</b>		
Establishes surveillance and monitoring of effluents, refer to Chapter 20, Subsection 20.11.3.		

### 2.9.1 Radiological Conditions in the Environment

To characterize the potential effects of the BWRX-300 operation on the surrounding environment, the baseline conditions must first be identified, described and delineated. Baseline radiation and radioactivity in the area of the DNNP site includes:

- Natural background
- Background from anthropogenic sources (fallout from nuclear testing and releases from other nuclear sites)
- Releases from activities on the Darlington Nuclear site, including operation of the existing DNGS, Tritium Removal Facility, and DWMF

Radiological emissions from the Darlington Nuclear site, including the DWMF, represented a small fraction of the DRLs. The four-year period 2016 – 2019 emissions ranged from 0.01 to

0.41% of the DRLs, as reported in the 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.9-1). The 2021 emissions were from 0.01 to 0.53% of the DRLs, as noted in the annual report on the results of the EMP N-REP-03443-10027 (Reference 2.9-2).

The radiological baseline conditions were established in the 2009 DNNP Environmental Impact Statement (EIS) – DNNP Environmental Assessment (EA) NK054-REP-07730-00029 (Reference 2.9-3). Updates to the radiological baseline conditions since the 2009 EIS-EA was conducted are discussed in detail in documentation including:

- The annual EMP report N-REP-03443-10027 (Reference 2.9-2)
- The 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.9-1)
- The 2020 DNNP – Site Preparation Licence Renewal Activity Report – Environment NK054-REP-01210-00110 (Reference 2.9-4)
- The 2020 DNNP Supporting Environment Studies – Environment NK054-REP-01210-0001 (Reference 2.9-5)
- The 2022 DNNP EIS NK054-REP-07730-00055 (Reference 2.9-16)

The 2020 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.9-4) concludes the radiological baseline conditions have not changed since the 2009 EIS-EA, per NK054-REP-07730-00029 (Reference 2.9-3). The same conclusion is reached in the 2022 DNNP EIS NK054-REP-07730-00055 (Reference 2.9-16). Details of these conditions are summarized in the following Subsections 2.9.1.1 and 2.9.1.2.

#### **2.9.1.1 Radiological Baseline Conditions**

The radiological baseline conditions in the area surrounding the Darlington Nuclear site are discussed in detail in the annual EMP report N-REP-03443-10027 (Reference 2.9-2), which demonstrates that all levels of radionuclides monitored around the Darlington Nuclear site remained stable since 2009 NK054-REP-07730-00029 (Reference 2.9-3). A Mann-Kendall trend analysis at the 95% confidence level did not indicate any statistically significant trends over the past 10 years for tritium in any medium sampled. For C-14, a Mann-Kendall trend analysis at the 95% confidence level over the past 10 years of data either indicated a statistically significant downward trend (C-14 in air at the Darlington Nuclear site boundary, C-14 in milk at dairy farms) or did not indicate any statistically significant trends (C-14 in fruit and vegetables, and C-14 in fish). A similar analysis was not conducted for noble gas parameters, as measurements taken at the Darlington Nuclear site boundary had average dose rates that were typically below detection limits.

Summaries are presented in the following paragraphs of the results of the annual results of the EMP report N-REP-03443-10027 (Reference 2.9-2), where sampling locations are available – as shown Figure 2.9-1.

#### **Air Samples**

Samples of air are collected to monitor the environment around the Darlington Nuclear site.

1. The 2021 tritium in air annual average concentrations measured at Darlington Nuclear site boundary locations ranged from 0.2 to 1.8 Bq/m<sup>3</sup>, with an average concentration of 0.87 Bq/m<sup>3</sup>. The 2021 annual average C-14 in air concentrations measured at Darlington Nuclear site boundary locations ranged from 206 to 248 Bq/kg-C, with an average concentration of 230 Bq/kg-C.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2. The annual boundary average noble gas dose rate is estimated from the monthly data from each detector. The Darlington Nuclear site boundary average dose rates for Ar-41, Xe-133, Xe-135, and Ir-192 are typically below the detection limits.

## **Terrestrial Samples**

Terrestrial baseline sampling is done in fruits and vegetables, milk, animal feed, eggs and poultry, and soil around the Darlington Nuclear site.

### **Fruits and Vegetables**

Fruits and vegetables, the 2021 average concentration for tritium near the Darlington Nuclear site was 17.8 Bq/L in fruits and 17.5 Bq/L in vegetables. The 2021 average concentration of C-14 was 230 Bq/kg-C in fruits and 248 Bq/kg-C in vegetables. A Mann-Kendall trend analysis of average fruit and vegetable activity at the 95% confidence level did not indicate any statistically significant trend over the past 10 years for tritiated water tritium and C-14.

### **Milk**

The 2021 average concentration of tritium was 4.3 Bq/L based on three dairy farms around the Darlington Nuclear site. The 2021 average concentration of C-14 in milk from dairy farm locations in the vicinity of the Darlington Nuclear site was 229 Bq/kg-C. A Mann-Kendall trend analysis of average milk activity at the 95% confidence level did not indicate any statistically significant trend over the past 10 years for tritium and C-14.

### **Animal Feed**

The average tritium concentration was 8.6 Bq/L for wet feed (forage). No dry feed samples were available in 2021. The average C-14 concentration in animal feed was 236 Bq/kg-C for wet feed (forage). No trend analysis was performed on animal feed since, beginning in 2013, wet feed and dry feed have been sampled separately, resulting in changes to sampling frequency and replicates.

### **Eggs and Poultry**

The concentration of tritium in eggs was 4.4 Bq/L and tritium in poultry was 10.3 Bq/L. Concentration of C-14 in eggs was 230 Bq/kg-C and in poultry was 229 Bq/kg-C. No trend analysis was performed as less than 10 years of data have been collected from sampling locations thus far.

### **Soil**

Soil is sampled every five years to identify possible radionuclide accumulation over time. The last soil sampling took place in 2017. Background values of Cs-137 are present in the soil as a result of historic weapons testing fallout. Co-60 and Cs-134, if detected, would be a result of emissions from the DNGS or other nuclear stations. In 2017, Cs-137 concentrations in background soil samples taken at provincial background locations ranged from 1.7 to 9.0 Bq/kg. All measured Cs-137 concentrations at locations around the Darlington Nuclear site in 2017 were within the range of values seen at the background locations, ranging from 5.1 to 7.2 Bq/kg. There is no indication of a buildup of activity in soil. Neither Cs-134 nor Co-60 were detected in any soil samples in 2017. Therefore, the Cs-137 measured in these soil samples is from historic weapons testing fallout and not from OPG Operations, as documented in the annual EMP report N-REP-03443-10017 (Reference 2.9-6).

## **Aquatic Samples**

Aquatic baseline sampling is done at nearby water supply plants, in well water, lake water, fish, and beach sand. As a result of the location of the Darlington Nuclear site, there are no depositional sediment locations near enough that are appropriate for sampling due to the high wave energy environment.

### Water Supply Plants

The impact of tritium emissions from OPG stations on the nearby water supply plants varies depending upon their distance from the station, lake current direction, location and depth of the water supply plant intake pipe as well as general dispersion conditions. Annual average tritium levels at all nearby water supply plants are well below the Ontario Drinking Water Quality Standard of 7,000 Bq/L. Annual average tritium concentrations measured at the Bowmanville, Newcastle, and Oshawa water supply plants in 2021 ranged from 4.6 to 6.6 Bq/L. Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium at any water supply plant near Darlington Nuclear site. Annual average gross beta activity levels at water supply plants were 0.11 Bq/L. This is well below the gross beta activity screening level of 1 Bq/L, which is a drinking water level recommended by Health Canada in the Guidelines for Canadian Drinking Water Quality: Guideline Technical Document.

### Well Water

The 2021 annual average tritium concentration observed in well water samples collected from the Darlington Nuclear site area was 12.0 Bq/L. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium in well water.

### Lake Water

The 2021 annual average tritium concentration observed in lake water samples collected from two beaches near the Darlington Nuclear site was 9.6 Bq/L. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level indicates no statistically significant trend for Darlington Nuclear site tritium in lake water.

### Fish

The 2021 tritium levels in the Darlington Nuclear site diffuser fish samples averaged <3.4 Bq/L, while the annual average C-14 level in same samples was 243 Bq/kg-C. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium or C-14 in Darlington Nuclear site fish. Cs-134 and Co-60, which are indicative of reactor operation, were not detected in any fish samples at Darlington Nuclear site in 2021. This is similar to past years. The average Cs-137 value for fish was 0.2 Bq/kg. The presence of Cs-137 in fish is primarily due to nuclear weapons testing and not reactor operation.

### Beach Sand

The average concentration of Cs-137 measured at beaches near the Darlington Nuclear site ranged from below detection (< 0.1) to 0.2 Bq/kg in 2021. Similar to previous years, there was no Co-60 or Cs-134 detected in any of the samples.

### Groundwater

In 2021, Darlington Nuclear site completed its annual groundwater monitoring program to evaluate groundwater quality and flow across the site and to detect any emergent issues. Tritium

concentrations at the sampled perimeter groundwater locations remained low. In general, tritium trends over time show that levels have remained nearly steady or decreased, indicating stable or improved environmental performance. There have been isolated cases within the DNGS protected area where tritium concentrations have shown increases, as reported in REP-07701-00001 (Reference 2.9-1). Where unexpected tritium concentrations are identified, investigations are completed to determine the root cause and to implement corrective measures. Ongoing results confirm that tritium in groundwater is mainly localized within the station protected area and the site perimeter tritium concentrations remain low.

### **2.9.1.2 Radiological Dose to the Public Due to Activities on DNGS Site**

The radiological public dose resulting from the operation of existing facilities on the Darlington Nuclear site is calculated annually and the results are published and made available to the public in the annual report summarizing the results of the EMP, per N-REP-03443-10027 (Reference 2.9-2). The dose calculations consider all significant pathways of exposure. Such calculations use the environmental pathway and dosimetric models and parameters that are provided in CSA N288.1-14 (Reference 2.9-7). The data used in the calculations consist of measurements of radionuclides released from the facility in environmental media obtained from the results of the yearly EMP report and consider background contributions where such data are available. For pathways or radionuclides where measured environmental data are not available, the dose is modelled from measured radionuclide emissions data reported in N-REP-03443-10027 (Reference 2.9-2).

Site public dose remains a small fraction of both the annual regulatory dose limit and annual natural background radiation in the area. The results of the annual EMP report N-REP-03443-10027 (Reference 2.9-2) conclude that the 2021 public dose for the Darlington Nuclear site was 0.6  $\mu\text{Sv}/\text{year}$  (represented by the adult farm resident critical group). The Darlington Nuclear site dose is <0.1% of the regulatory limit of 1,000  $\mu\text{Sv}/\text{year}$  for a member of the public, and <0.1% of the background radiation around Darlington Nuclear site. As can be seen in the 2016-2021 EMP reports, the 2016 to 2021 public dose estimates for the critical groups are at most approximately 0.08% of the regulatory public dose limit of 1,000  $\mu\text{Sv}/\text{year}$ , and at most approximately 0.06% of the dose from background radiation (1.4 mSv/year) in the vicinity of Darlington Nuclear site.

The public dose is also reported in the Darlington Nuclear site ERA, which is routinely updated in accordance with REGDOC-3.1.1 (Reference 2.9-8). A CSA N288.6-12 (Reference 2.9-9) compliant ERA was produced for the Darlington Nuclear site in 2020 D-REP-07701-00001 (Reference 2.9-1) and included a human health risk assessment and ecological risk assessment for both radiological and non-radiological parameters and physical stressors. The ERA concluded that the Darlington Nuclear site is operating in a manner that is protective of human and ecological receptors residing in the surrounding area. No discernable health effects are anticipated due to the exposure of potential critical groups to the radiological effluent from the Darlington Nuclear site. Demonstration that the critical groups are protected implies that other receptor groups near the Darlington Nuclear site are also protected.

### **2.9.2 Radiation Monitoring Systems**

OPG's radiation monitoring systems, which are currently used for DNGS, comprise on-site, site boundary, and off-site monitoring systems. Detailed information about environmental sampling locations, sampling frequency, the number of samples taken, the media sampled, the sampling method, and the radionuclides monitored can be found in CSA N288.4 on Environmental Monitoring Programs at Nuclear Facilities and Uranium Mines and Mills (Reference 2.9-10). Summaries of four specific aspects of the radiation monitoring systems are presented as follow:

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

1. Environmental monitoring systems, including the environmental off-site and site boundary monitoring as well as samples taking and analysis – Subsection 2.9.2.1
2. The off-site and site boundary TLD sites – Subsection 2.9.2.2
3. The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary – Subsection 2.9.2.3
4. The site Effluent Monitoring Program – Subsection 2.9.2.4

#### **2.9.2.1 Environmental Monitoring Program**

The environmental monitoring systems and sampling programs detailed in the annual EMP report N-REP-03443-10027 (Reference 2.9-2) include off-site and site boundary monitoring and are summarized here. Samples taken are analysed at certified laboratories or laboratories with documented comprehensive quality assurance and quality control programs, in accordance with clause 8.3.2 of CSA N288.4 (Reference 2.9-10). The Canadian Association for Laboratory Accreditation certified OPG Health Physics Laboratory, and external contractors, perform the sample collection and analysis for Darlington Nuclear site and provincial EMPs, as per N-PROC-OP-0025 R012 (Reference 2.9-11). Sampling locations are shown in Figure C1 in Appendix C of N-REP-03443-10027 (Reference 2.9-2), which is replicated in Figure 2.9-1.

##### **2.9.2.1.1 Atmospheric Sampling**

Concentrations in air are sampled to monitor the environment around the Darlington Nuclear site. Tritium, C-14, and noble gases are measured and reported in N-REP-03443-10027 (Reference 2.9-2).

1. The active tritium in air sampler collects water vapor by passing air continuously at a steady rate through two molecular sieve canisters in series. The active samplers are located at six site boundary EMP monitoring locations around the Darlington Nuclear site. These samples are collected and analysed monthly.
2. C-14 in air is sampled using passive sampling technology. The passive C-14 sampler works by absorption of CO<sub>2</sub> in air into soda lime pellets exposed for a period of an annual quarter. Samples are analysed after each quarter. C-14 in air is monitored at four boundary locations for the Darlington Nuclear site.
3. External gamma radiation doses from noble gases and Ir-192 are measured using sodium iodide (NaI) spectrometers set up around the Darlington Nuclear site. There are a total of eight detectors around the Darlington Nuclear site that monitor the dose rate continuously.

##### **2.9.2.1.2 Aquatic Sampling**

Samples of drinking water sources (municipal and well water), lake water, beach sand and fish are collected to monitor the aquatic environment around the Darlington Nuclear site. Tritium, gross beta, C-14, and gamma activity are measured and reported in N-REP-03443-10027 (Reference 2.9-2).

1. Samples of drinking water are taken during each 8-12-hour shift at water supply plants that supply water to Durham Region the Bowmanville water supply plant, the Newcastle water supply plant, and the Oshawa water supply plant. Weekly composites of these samples are analysed for tritium, and monthly composites are analysed for gross beta activity.
2. Monthly well water samples are collected from four wells around the Darlington Nuclear site area. Samples are analysed monthly for tritium.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

3. Lake water for recreational use is sampled from two beaches in the vicinity of the Darlington Nuclear site on a monthly basis and analysed for tritium. It is used to assess the water immersion dose exposure pathway from swimming in lake water.
4. At the Darlington Nuclear site, fish sampling takes place over the cooling water discharge diffuser. The target fish species to be collected at Darlington Nuclear site and at background locations is White Sucker, with Brown Bullhead as the backup species. Eight replicate fish samples are collected and analysed at each location. A sample consists of the fish muscle tissue only, and excludes the head, skin, fins, and as many bones as possible. Tritium, C-14, Co-60, Cs-134, Cs-137, and Potassium-40 (K-40) measurements are performed on each fish sample.
5. Sand from three beaches around the Darlington Nuclear site is collected annually to represent a potential pathway for external dose. Eight replicates are collected per sampling location. Gamma spectrometry is performed on these samples. Beach sand samples were collected at a background location to determine the Cs-137 concentrations in sand due to atmospheric weapons test fallout.
6. Groundwater monitoring occurs of each year, with 81 groundwater monitoring locations at Darlington Nuclear site sampled in 2021 for tritium, the key parameter of concern, refer to NK38-REP-10140-10031 (Reference 2.9-12). Annual water level measurements are also conducted.

#### **2.9.2.1.3 Terrestrial Sampling**

Samples of soil, fruits, vegetables, animal feed, milk, eggs, and poultry are collected to support the public dose calculation for the Darlington Nuclear site. Terrestrial biotas receive exposure from both airborne and waterborne emissions. Tritium and C-14 are measured, per N-REP-03443-10027 (Reference 2.9-2).

1. Fruits and vegetables are sampled three times from each location for a representation of the entire growing season. Each sample is analysed for C-14 and tritium. A total of five locations for fruit and vegetable were sampled around the Darlington Nuclear site.
2. Milk sampling is used to estimate the portion of dose received from milk ingestion for the dairy farm potential critical group. Milk samples are collected on a monthly basis from dairy farms around the Darlington Nuclear site and analysed for tritium and C-14. Samples are collected from three dairy farms around the Darlington Nuclear site.
3. Locally grown animal feed is collected from four dairy farms around the Darlington Nuclear site, twice a year, with two replicates collected per visit. Since 2013, dry feed (grains, hay, etc.) and wet feed (forage) are collected separately. Animal feed is analysed for tritium and C-14.
4. Eggs are sampled on a quarterly basis and three sample replicates are collected per visit. Poultry is collected annually with eight sample replicates collected per visit. Both eggs and poultry are analysed for tritium and C-14. One farm location around the Darlington Nuclear site is sampled for eggs.

#### **2.9.2.2 Thermoluminescent Dosimeter Monitoring**

TLDs are located around the Darlington Nuclear site perimeter as well as at off-site locations. The TLDs contain field cards that passively monitor the airborne dose over the course of a year. Cards are read and analysed annually when they are changed. The net readings for the four elements from the field card readings are input to an algorithm that converts the readings into air kerma (short for Kinetic Energy Released per unit mass of Air, which is a measure of energy in

joules (J) deposited in a unit mass (kg) of air; thus, in J/kg), ambient dose equivalent and directional dose equivalent, as described in N-PROG-RA-0001 (Reference 2.9-13).

Also, TLDs are located around the DWMF fence line. The DWMF perimeter dose rates are measured and reported quarterly.

#### **2.9.2.3 Gamma Monitoring System**

The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary, is a fixed radiological detection and monitoring system designed to provide real-time gamma dose rate measurements, as reported in N-PROG-RA-0001 (Reference 2.9-13). Refer to Chapter 19, Section 19.3 for additional relevant information.

#### **2.9.2.4 Effluent Monitoring Program**

The Darlington Nuclear Site Effluent Monitoring Program is governed by OPG's N-STD-OP-0031 Monitoring of Nuclear and Hazardous Substances in Effluents (Reference 2.9-14). This standard establishes minimum requirements to establish an appropriate surveillance and monitoring program for nuclear and hazardous substances in airborne and waterborne effluents from operating OPG Nuclear facilities, including the DNGS, in accordance with CSA N288.5-11 (Reference 2.9-15). The effluent monitoring program is further discussed in Chapter 20, Subsection 20.11.3.



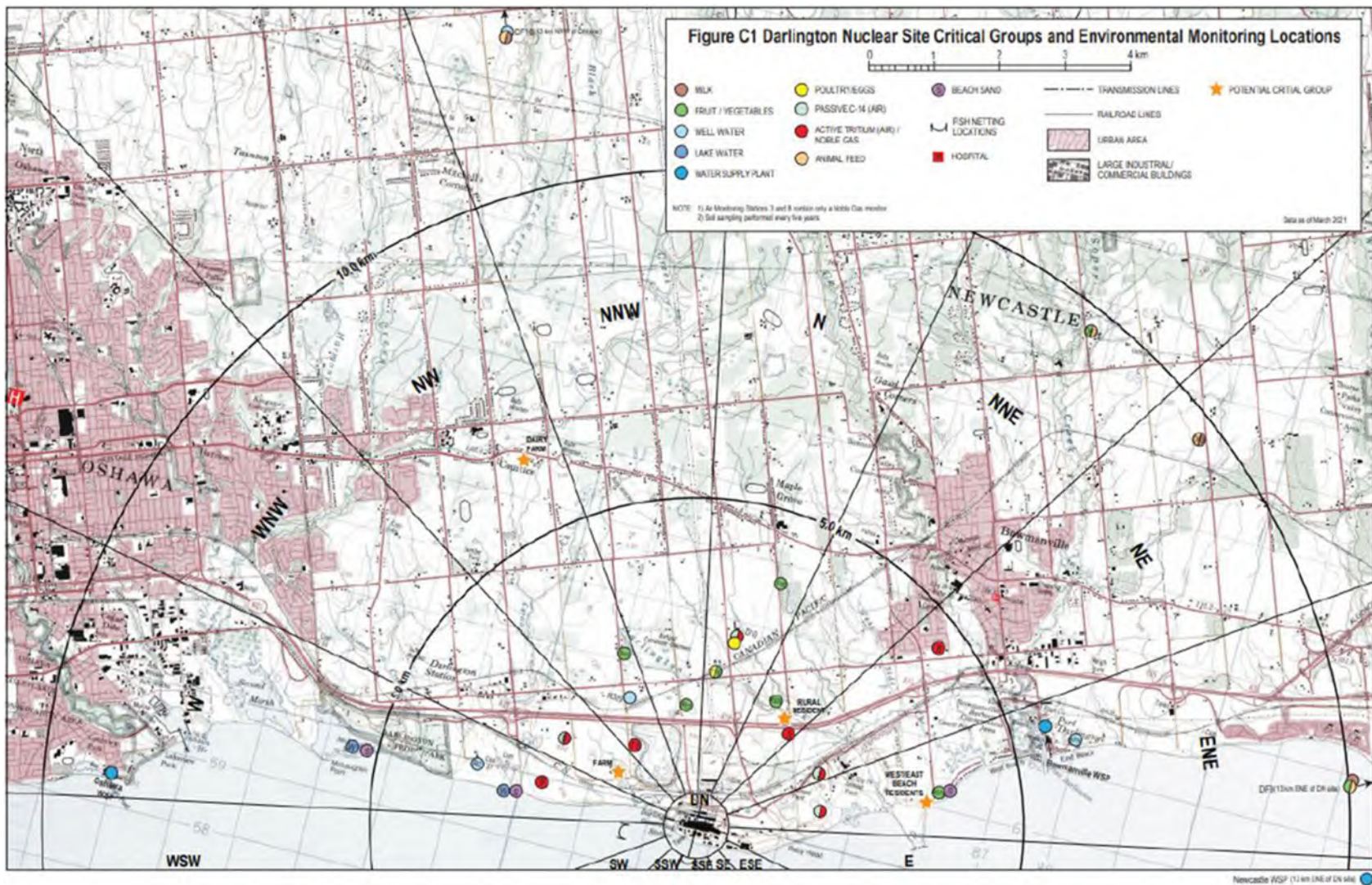


Figure 2.9-1: Darlington Nuclear Site Critical Groups and Environmental Monitoring Locations (Reference 2.9-2)

### 2.9.3 References

- 2.9-1 D-REP-07701-00001 R001, 2020, "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.9-2 N-REP-03443-10027 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-3 NK054-REP-07730-00029 R000, 2009, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.9-4 NK054-REP-01210-00110 R001, 2020, "DNBP – Site Preparation Licence Renewal Activity Report – Environment," Ontario Power Generation.
- 2.9-5 NK054-REP-01210-0001 R000, 2020, "Darlington New Nuclear Project Supporting Environment Studies – Environment," Ontario Power Generation.
- 2.9-6 N-REP-03443-10017 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-7 CSA N288.1-14 "Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities," CSA Group.
- 2.9-8 CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants."
- 2.9-9 CSA N288.6-12, "Environmental Risk Assessments At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-10 CSA N288.4, "Environmental Monitoring Programs At Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-11 N-PROC-OP-0025 R012, "Management of the Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-12 NK38-REP-10140-10031 R001, 2021, "Darlington Nuclear Groundwater Monitoring Program Results," Ontario Power Generation.
- 2.9-13 N-PROG-RA-0001 R019, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.9-14 N-STD-OP-0031 R009, "Monitoring of Nuclear and Hazardous Substances in Effluents," Ontario Power Generation.
- 2.9-15 CSA N288.5-11, "Effluent Monitoring Programs At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-16 NK054-REP-07730-00055 R000 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.



## 2.10 Site-Related Issues in Emergency Preparedness and Response and Accident Management

The information presented in Section 2.10 includes:

- General Consideration – Subsection 2.10.1
- Feasibility of Emergency Preparedness and Response – Subsection 2.10.2
- Evacuation Time Estimates and Route – Subsection 2.10.3
- Support Networks in the Vicinity of the Site – Subsection 2.10.4
- Administrative Measures with External Organizations – Subsection 2.10.5

In Table 2.10-1, a summary description is included of site-related emergency preparedness and response feasibility, relevant evacuation time estimates; supporting agencies and services; communication systems; provincial and on-site plans; and other nuclear organization.

**Table 2.10-1: Summary of DNNP Site Relevant Characteristics and Parameters**

Characteristic	Value/Description		
2.10.2 Feasibility of Emergency Preparedness and Response			
Accessibility	<ul style="list-style-type: none"><li>• Studies considered number of personnel on site, regional population change, infrastructure updates, geography, and weather patterns.</li><li>• Main entrance: Holt Road South via Energy Drive, or Highway 401, or Park Road via Highway 401 to Energy Road.</li></ul>		
DNNP Traffic Management Plan	Developed to guide site transportation demands during various phases of project, including construction		
BWRX-300 Design	<ul style="list-style-type: none"><li>• Incorporates reliable and passive safety functions with redundancy and diversity that satisfy safety goal requirements</li><li>• Informed by DSA and Probabilistic Safety Assessment (PSA) results to develop optimized accident management strategies and measures.</li></ul>		
2.10.3 Evacuation Time Estimates and Route			
Estimates	<ul style="list-style-type: none"><li>• Provides off-site emergency planners with projections on how long it may take for various emergency planning sectors and the Detailed Planning Zone (DPZ) to evacuate.</li><li>• Considered various scenarios as time of day, day of week, road restrictions, special event assemblies and weather conditions.</li></ul>		
Routes	<ul style="list-style-type: none"><li>• On-site process and travel route for site evacuations are documented in site-specific instructions, including DNNP site during various phases of the project.</li><li>• Measures to evacuate publicly accessible areas on the Darlington Nuclear site.</li></ul>		
Infrastructure	Impacted local businesses and transportation networks		
2.10.4 Support Networks in the Vicinity of the Site			
Agencies, Businesses, Services, Plans	<ul style="list-style-type: none"><li>• Ambulances and Hospital</li><li>• Municipal services</li><li>• Potassium Iodide Program</li></ul>	<ul style="list-style-type: none"><li>• Police force</li><li>• Alerting systems</li><li>• PNERP</li><li>• Consolidated Nuclear Emergency Plan (CNEP)</li></ul>	<ul style="list-style-type: none"><li>• On-site and off-site communication systems</li><li>• Information to media</li></ul>

**Table 2.10-1: Summary of DNNP Site Relevant Characteristics and Parameters**

Characteristic	Value/Description
Off-site Alerting System	Managed by Durham Region and the Province of Ontario
Designated and Host Municipalities	<ul style="list-style-type: none"> <li>Administered the Potassium Iodide Program</li> <li>Provide centres for Emergency Workers, Evacuation, and Reception (with personnel and resources support provided from OPG)</li> </ul>
<b>2.10.5 Administrative Measures with External Organizations</b>	
The Province of Ontario, Provincial Nuclear Emergency Response Plan (PNERP)	<ul style="list-style-type: none"> <li>Provides the off-site planning basis for nuclear emergencies with the goal of ensuring public safety in the event of a nuclear emergency</li> <li>Establishes the principles, concepts, organization, responsibilities, policy, functions, and interrelationships which govern all off-site nuclear emergency planning, preparation, and response in Ontario</li> </ul>
Other Nuclear Partners	<ul style="list-style-type: none"> <li>Nuclear partners in Canada are expected to respond, if necessary</li> <li>CANada Deuterium Uranium (CANDU) Owners Group for support and technical assistance</li> <li>Institute of Nuclear Power Operations (INPO) for necessary support from the industry</li> </ul>

### 2.10.1 General Consideration

In accordance with Subsection 4.10.2 of REGDOC-1.1.2 (Reference 2.10-17), OPG, as the licensee for the BWRX-300 facility, has established an effective DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1) which is governed by OPG CNEP N-PROG-RA-0001 (Reference 2.10-2). These two plans cover aspects such as:

- Feasibility of emergency preparedness and response
- Local infrastructure for evacuation adequacy
- Availability of support networks in the vicinity of the site
- Availability of transport, communication and infrastructure external to site
- Need for administrative measures
- Roles of response organization other than OPG

Elaboration on these aspects and associated detailed information are included in the following Subsections 2.10.2 to 2.10.5.

### 2.10.2 Feasibility of Emergency Preparedness and Response

The BWRX-300 facility accessibility for OPG personnel, contractors, and response crews, as well as for the transport of any equipment necessary in an emergency, is critical for the purposes of emergency preparedness and response at the DNNP site. Such accessibility is considered by OPG in the design of the BWRX-300 facility for the construction, commissioning, operation, and decommissioning phases. In this regard, events at both the DNNP site and the existing DNGS site are considered since an event at one site may affect personnel and the emergency response at the other site. Emergency response is, therefore, considered for the entire Darlington Nuclear site. Protocols throughout the project phases are included in the DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1).

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

To ensure accessibility for both off-site emergency responders and on-site personnel to and from the DNNP site, OPG conducted studies that considered estimated number of all personnel in the Darlington Nuclear site, regional population changes, infrastructure updates, geography, and weather patterns. The results of these studies are formalized into plans and reports to assist with emergency planning; primarily in, DNGS Development of Evacuation Time Estimates, per NK38-REP-03490-10133 (Reference 2.10-3), Summary Report: Site Evaluation Studies for Nuclear Installations at Darlington External Human Induced Events NK054-REP-01210-00010 (Reference 2.10-4), Darlington New Nuclear Project Traffic Management Plan NK054-PLAN-08965.4-00001 (Reference 2.10-5) and Updated Traffic Management Plan NK054-REP-07730-0969014 (Reference 2.10-20). In addition, detailed analysis of the DNNP site accessibility is noted in Site Evaluation for OPG New Nuclear at Darlington – Nuclear Safety Considerations NK054-REP-01210-00008 (Reference 2.10-6).

The main entrance to the DNNP site is per the existing entrance to the entire Darlington Nuclear site via Holt Road South in Bowmanville, Ontario. Holt Road South is accessible via Energy Drive eastbound on Highway 401 and has a direct off-exit of Highway 401 westbound. An alternate access point from westbound Highway 401 to Energy Drive is Park Road. Park Road traverses the western part of the Darlington Nuclear site, crossing 2<sup>nd</sup> Line, which then connects to Holt Road. Energy Drive west of Park Road is named Megawatt Drive. Additional detailed information on transportation networks on the Darlington Nuclear site and in the surrounding area is provided in Subsection 2.1.5.

For the purpose of Subsection 2.10.2, a generic site map displaying the Darlington Nuclear site in relation to major roadways is shown in Figure 2.10.2-1, where the area allotted to DNNP is shaded in yellow east of the DNGS area, and the DNGS exclusion zone of 914 m is shown, per D-PLAN-00120-0001 (Reference 2.10-7).

The DNNP Traffic Management Plan (Reference 2.10-5) was initiated by OPG to guide, in a safe manner, site transportation demands during various phases of the BWRX-300 facility including construction. This Traffic Management Plan assesses the impact of traffic within the vicinity of the DNNP site, in the area noted in Figure 2.10.2-2.

Chapter 15, Subsection 15.6.1 states that the specific objectives of the PSA and severe accident analysis (SAA) are to demonstrate that the BWRX-300 facility is designed with features that incorporate highly reliable and available passive safety functions with significant redundancy and diversity to comply with the safety goal requirements in REGDOC-2.5.2 (Reference 2.10-9).

Further, as described in Chapter 15, Subsection 15.1.5, DEC's are identified to aid in designing and implementing safety features (complementary design features) to mitigate the consequences of DEC's. The Severe Accident Management (SAM) program is informed by the insights of the Deterministic Safety Analysis (DSA) and results of the PSA for the development, implementation, training and optimization of accident management strategies and measures, as identified in Chapter 15 Subsection 15.6.1.

Additionally, Chapter 13, Subsection 13.4.3 discusses the programmatic approach to develop emergency operating procedures and severe accident management guidelines (SAMG) in accordance with REGDOC-2.3.2 (Reference 2.10-21).

### **2.10.3 Evacuation Time Estimates and Route**

OPG made available to off-site planning authorities a revised Darlington Site Evacuation Time Estimate, per NK38-REP-03490-10133 (Reference 2.10.3) using the 2016 National Census Data with per decade population projections out to 2088, as well as current and forecasted infrastructure.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

The estimate provides off-site emergency planners with projections on how long it may take for current emergency planning sectors and the DPZ to evacuate if required. Variables such as time of day, day of week, road restrictions, special event assemblies and weather were assessed as to how those factors may impact the evacuation duration. In the first quarter of 2023, OPG will issue an updated Darlington Site Evacuation Time Estimate based on the 2021 national census data and will subsequently be shared with stakeholders.

On-site, the process and travel route for site evacuations are described in D-INS-0349-10030 (Reference 2.10-10). The current revision of such OPG's instructions considers the DNNP site during various phases of the project. The main exit routes are via:

1. Park Road to Energy Drive to Highway 401 westbound
2. Old Holt Road, continuing onto Holt Road northbound of Highway 401 east and westbound

During an evacuation from the Darlington Nuclear site, Energy Drive will be closed, as necessary, by local police between Park Road to Holt Road to control traffic volume and delays. Additionally, procedures exist for OPG to evacuate publicly accessible areas on the Darlington Nuclear site, per INS-03490-10015 (Reference 2.10-11), including the Darlington Waterfront Trail and the Hydro Soccer Fields (refer to Subsections 2.1.7 and 2.1.8).

Local infrastructure within the vicinity of the DNNP site is described in Section 2.3, which includes local businesses, and transportation networks that are impacted by an emergency on-site in their current and future expanded state.

#### **2.10.4 Support Networks in the Vicinity of the Site**

Collaboration of OPG with local government agencies and businesses is essential to the DNNP emergency response capabilities. Shared roads, emergency services, communication networks, and transportation networks are utilized to assist with site response, evacuation, and relocation services, as required.

During construction, prior to turnover to Operations, the fire protection controls and response are primarily the responsibility of the prime contractor or constructor, per CSA N293-12 (Reference 2.10-18). Once handover to Operations occurs, OPG's own fire protection program, with its necessary updates for the BWRX-300 facility, will be in place and be compliant with CSA N293S1:21 (Reference 2.10-19).

Arrangements also exist for local ambulance service and hospital support for casualties from the Darlington Nuclear site. Toronto Hospital Corporation, Western Division, has been provincially designated and funded as the radiation trauma centre for Ontario. This includes the capability to deal with contaminated casualties, trauma, and acute radiation syndrome. Lakeridge Health—Bowmanville Hospital is the primary local hospital designated to receive contaminated casualties from DNGS. DNNP is expected to be included in this agreement, encompassed under the Darlington Nuclear site. Agreements are also in place to provide support to the site from the local police force in the event of an on-site security event (Reference 2.10-2). Subsection 2.1.4 and Subsection 2.1.6 provide, respectively, additional details on Municipal Services local to the area as well as on public transit.

To communicate with off-site emergency responders during an event, OPG currently uses Durham NEXGEN P-25 Radio system – part of the Durham Emergency Communication. As the DNNP progresses, and prior to Operations, these systems will be assessed for future use.

As noted in Chapter 9A, Subsection 9A.9.1.3, the off-site communication system is designed to satisfy emergency plan requirements for accident conditions, including notification of personnel and implementation of evacuation procedures. This capability includes communications support

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

to both on-site and off-site emergency response facilities; at least one on-site and one off-site communications system, each with a backup power source. The on-site communications involve immediate notification process and secondary communication methods to alert all on-site personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels. This capability also includes communications support for firefighting, including support of alternative and dedicated shutdown capabilities.

As noted in Subsection 2.10.2, the SAMG is informed by the insights of the DSA and results of the PSA for the development, implementation, training and optimization of accident management strategies and measures. This includes review of the existing Beyond Design Basis Accident telecommunications equipment designated for DNGS, which also are considered for DNNP and rely on external infrastructure to function.

Durham Region and the Province of Ontario manage alerting systems to let the public know when a nuclear emergency occurs. Durham Region's public alerting system includes loud sirens, located within the Automatic Action Zone of the Darlington Nuclear site and an automated landline telephone calling system. The automated telephone system sends a recorded message to landline phones in the DPZ area around the nuclear station. The Province of Ontario manages the Alert Ready system. These alerts broadcast through television, radio, and cellphones. The off-site public alerting systems are currently applicable to DNGS but expected to be utilized for DNNP. Prior to fuel-in commissioning, this will be identified as part of the revised PNERP (Reference 2.10-12).

The DNGS station has an established Potassium Iodide Program, which satisfies the requirements of the PNERP (Reference 2.10-12) and REGDOC-2.10.1 (Reference 2.10-13), both are encompassed by the CNEP (Reference 2.10-2). The program is supported by designated municipalities to ensure continued availability of Potassium Iodide to residents of the DPZ and Ingestion Planning Zone, and information is available to the general public, including on-line, as per N-GUID-03491-10011 (Reference 2.10-14). Similar to the public alerting systems, this program is currently applicable to DNGS, but expected to be utilized for DNNP. Prior to fuel-in commissioning, this will be identified as part of the revised PNERP.

The PNERP (reference 2.10-12) outlines the requirements for designated municipalities and host municipalities to include provisions for Emergency Worker Centres, Evacuation Centres, and Reception Centres in the unlikely event of an evacuation, as noted in D-INS-0349-10030 (Reference 2.10-10). OPG supports these Off-site Centres by providing personnel and resources for personal monitoring and decontamination. The current facilities applicable to the DNGS are listed in Appendix C3.4 of CNEP (Reference 2.10-12). It is to be determined whether such facilities are required for DNNP which, if so, will be reflected in a future revision of the PNERP. Additionally, OPG has two Mobile Monitoring and Decontamination Units that are poised and ready for deployment when designated by the Provincial Emergency Operations Centre (PEOC). OPG deploys on-site and off-site radiation survey teams to the area, if required.

The Joint (Emergency) Information Centres intending to disseminate Information to the media are also set up between OPG, the Province of Ontario, and local municipalities. Refer to the CNEP (Reference 2.10-2). OPG's Nuclear Crisis Communication Standard (Reference 2.10-15) provides corporate direction for assisting with site emergencies. This standard outlines how information is passed between the incident station, emergency response facilities, Corporate Media Desk, and the public domain.

There are no known issues at this time that would hinder the implementation of DNNP emergency response actions. OPG is currently working with the Province of Ontario to develop timelines for PNERP revisions to incorporate a separate implementing plan for the DNNP site or as part of the DNGS site implementing plan.

Descriptions of the development of the DNNP emergency response plan, and the emergency response facilities are detailed in Chapter 19, Sections 19.1 and 19.2, respectively.

#### **2.10.5 Administrative Measures with External Organizations**

In the Province of Ontario, Canada, the PNERP (Reference 2.10-12) provides the off-site planning basis for nuclear emergencies with the goal of ensuring public safety in the event of a nuclear emergency. The PNERP Master Plan (Reference 2.10.12) establishes the principles, concepts, organization, responsibilities, policy, functions, and interrelationships which govern all off-site nuclear emergency planning, preparation, and response in Ontario. Each nuclear facility identified in the PNERP has its own implementing plan which is site-specific in nature and deals with local characteristics, planning and operational particulars. OPG has a memorandum of understanding in place with the Province of Ontario to revise the PNERP prior to fuel-in commissioning to include DNNP and issue a revised Darlington implementing plan or a separate implementing plan for DNNP (Reference 2.10-1).

OPG continues to collaborate with the Province of Ontario and other external organizations responsible for off-site nuclear emergency planning to ensure the implementation of their respective emergency plans and related protective actions accommodate the lifecycle of BWRX-300 facility built on the DNNP site.

Other nuclear partners within Canada are requested to respond where necessary, for any assistance in a nuclear event at DNGS and DNNP, as per the existing mutual aid response memoranda of understanding.

OPG also has arrangements for support and technical assistance with the CANDU Owners Group members and INPO, a consortium of nuclear utilities, to obtain any necessary support available from the industry during an emergency. INPO operates a 24-hour emergency assistance line and an Emergency Response Centre in Atlanta, Georgia, USA, to provide support to member utilities.

Further information on external administrative assistance is provided in the Emergency Planning and Preparedness Technical Support Document: New Nuclear – Darlington Environmental Assessment NK054-REP-07730-00021 (Reference 2.10-8), and the DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1).





Figure 2.10.2-1: Darlington Nuclear Site Showing DNGS and DNNP Areas





Figure 2.10.2-2: Area of Consideration for Traffic Management Plan



### 2.10.6 References

- 2.10-1 NK054-PLAN-01210-00002, "DNNP Nuclear Emergency Preparedness Plan," Ontario Power Generation.
- 2.10-2 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.10-3 NK38-REP-03490-10133, "Darlington NGS Development of Evacuation Time Estimates," Ontario Power Generation.
- 2.10-4 NK054-REP-01210-00010, "Summary Report: Site Evaluation Studies for Nuclear Installations at Darlington External Human Induced Events," Ontario Power Generation.
- 2.10-5 NK054-PLAN-08965.4-00001, "Darlington New Nuclear Project Traffic Management Plan (TMP)," Ontario Power Generation.
- 2.10-6 NK054-REP-01210-00008, "Site Evacuation for OPG new Nuclear at Darlington – New Nuclear Safety Considerations," Ontario Power Generation.
- 2.10-7 D-PLAN-00120-0001, "Darlington Nuclear Generating Station Campus Plan," Ontario Power Generation.
- 2.10-8 NK054-REP-07730-00021, "Emergency Planning and Preparedness Technical Support Document: New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.10-9 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities; Nuclear Power Plants."
- 2.10-10 D-INS-03490-10030, "Evacuation Relocation," Ontario Power Generation.
- 2.10-11 D-INS-03490-10015, "Security First Line Manager," Ontario Power Generation.
- 2.10-12 Ontario Provincial Nuclear Emergency Response Plan (PNERP) – Master Plan, 2017.
- 2.10-13 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 2.10-14 N-GUID-03491-10011, "Potassium Iodide (KI) Pill Administration Guide," Ontario Power Generation.
- 2.10-15 N-STD-AS-0010, "Nuclear Crisis Communication Standard," Ontario Power Generation.
- 2.10-16 OPG-PROC-0028, "Crisis Management and Communications Centre Procedure," Ontario Power Generation.
- 2.10-17 CNSC Regulatory Document REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.10-18 CSA N293-12, "Fire Protection for Nuclear Power Plants," CSA Group.
- 2.10-19 CSA N293S1:21, "Supplement No. 1 to N293-12, Fire protection for nuclear power plants (application to small modular reactors)," CSA Group.
- 2.10-20 NK054-REP-07730-0969014, "Updated Darlington New Nuclear Project Traffic Management Plan (TMP)," Ontario Power Generation.
- 2.10-21 CNSC Regulatory Document REGDOC-2.3.2, "Operating Performance – Accident Management."

## 2.11 Monitoring of Site-Related Parameters

Section 2.11 provides a description of the strategy for monitoring site-related parameters relevant to the DNNP site, with emphasis on the site parameters that need to be monitored for the hazards identified in Section 2.2 which affect the DNNP through the lifecycle of the BWRX-300 facility. The information in Section 2.11 satisfies the requirements of Subsection 4.5.2 of REGDOC-1.1.2 (Reference 2.11-15) and the guidance of Subsection 7.4.2 of REGDOC-2.5.2 (Reference 2.11-16).

The information in Section 2.11 covers:

- Volcanic Phenomena Monitoring – Subsection 2.11.1
- Surface Faulting Monitoring – Subsection 2.11.2
- Seismic and Geotechnical Monitoring – Subsection 2.11.3
- Meteorological Monitoring – Subsection 2.11.4
- Hydrological monitoring – Subsection 2.11.5
- Radiation Monitoring – Subsection 2.11.6
- Environmental Monitoring – Subsection 2.11.7
- Biological Organisms and Human Induced Hazards Monitoring – Subsection 2.11.8
- Long Term Monitoring Program – Subsection 2.11.9

Table 2.11-1 summarizes key DNNP characteristics and the approach for monitoring key site parameters.

**Table 2.11-1: DNNP Site Characteristics and Parameters Monitoring Approach**

Characteristic	Monitoring Approach
2.11.1 Volcanoes Monitoring	Hazard Screened out – No site-specific parameter to be monitored
2.11.2 Surface Faulting Monitoring	Hazard Screened out – No site-specific parameter to be monitored. Any changes will be evaluated within the long-term monitoring program.
2.11.3 Seismic and Geotechnical Monitoring	<ul style="list-style-type: none"> <li>• Southern Ontario Seismic Network stations on Darlington Nuclear site</li> <li>• Current site-specific information is used during construction, with monitoring of excavation and blasting effects.</li> <li>• The Foundation Interface Analysis (FIA) work in (Reference 2.11.19) is fed by the site-specific parameters reported in (Reference 2.11-20) and will be updated by monitored specific geotechnical and seismic parameters during operation.</li> <li>• In-service monitoring approach of and instrumentation for BWRX-300 structures include testing and surveillance programs for below-grade structures and foundations over their design lives</li> </ul> <p>Field instrumentation system with recordings is benchmarked against design estimates of settlement and vertical and horizontal movement around the deeply embedded RB and the foundations of the Control Building (CB), TB, and RWB</p>

**Table 2.11-1: DNNP Site Characteristics and Parameters Monitoring Approach**

Characteristic	Monitoring Approach	
2.11.4 Meteorological Monitoring	<ul style="list-style-type: none"> <li>On-site meteorological tower</li> </ul> Environment Canada maintained stations, and notification on severe weather conditions	
2.11.5 Hydrological Monitoring	<ul style="list-style-type: none"> <li>Precipitation, groundwater flow and groundwater hydrology</li> <li>Lake Ontario water levels</li> </ul> Lake current real-time monitoring system	
2.11.6 Radiation Monitoring (refer to Section 2.9)	<ul style="list-style-type: none"> <li>Environmental off-site and site boundary monitoring and sampling</li> <li>Off-site and site boundary TLD sites</li> <li>Automated Gamma monitoring system</li> </ul> Effluent Monitoring Program	
2.11.7 Environmental Monitoring	Environmental Monitoring Program, detailed in Chapter 20, Subsection 20.11.2	
2.11.8 Biological Organisms and Human Induced Hazards Monitoring	Waterborne, and Airborne Hazards and Biological Organisms	Monitored and controlled in a manner to enable the continued safe operation of the BWRX-300
	Human Induced Hazards—General	Screened out based on Design Mitigation – No Site-specific parameter to be monitored
	Air Transportation activities	Hazard Screened out – No site-specific parameter to be monitored
	Chemical Explosions	Screened out based on Design Mitigation – No Site-specific parameter to be monitored
	Activities at nearby industrial and other facilities	St. Marys Cement plant seismic monitoring station
2.11.9 Long Term Monitoring Program	To be determined potential impacts of climate changes on BWRX-300 operation via long-term monitoring, review, and updates	

### 2.11.1 Volcanic Phenomena Monitoring

There are no volcanic structures or active volcanoes in the vicinity of the DNNP site. Therefore, the volcanic hazard is not a potential hazard to the DNNP site, and no site-specific parameter to be monitored for this hazard as it is screened out, as per the 2020 DNNP application to renew the Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.11-2).

### 2.11.2 Surface Faulting Monitoring

There are no active surface faults or tectonic plates in the vicinity of the DNNP site. Therefore, there is no site-specific parameter to be monitored for surface faulting hazard at the DNNP site as this is screened out, as described in the 2020 NK054-CORR-00531-10533 (Reference 2.11-2). Any changes in this hazard are to be evaluated as part of the long-term monitoring program.

### 2.11.3 Seismic and Geotechnical Monitoring

Site-related parameters are monitored to account for effects from seismic or geotechnical hazards, including earthquakes. Characterization of the seismicity of the region surrounding the

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

site, using the Southern Ontario Seismic Network stations on Darlington Nuclear site, forms an essential part of the assessment of the seismic hazard.

Considering the proximity of the DNNP and DNGS sites, the updated hazard curve characterizing the seismic conditions for DNGS in the 2021 Darlington Risk Assessment NK38-REP-03611-10041 (Reference 2.11-1) is deemed applicable to the DNNP site and, thus, is to be utilized during the design and construction stages of the BWRX-300 facility.

The DNNP site-specific geotechnical considerations are discussed in Section 9.3 of the 2009 DNNP Site Evaluation of geotechnical aspects NK054-REP-01210-00011 (Reference 2.11-3). During the construction of the BWRX-300 facility, the effects of any excavation or blasting is to be monitored for their impact on the existing DNGS Power Blocks.

All permanent cut/fill slopes within the areas for DNNP site are to be instrumented and monitored regularly during and after completion of construction and during operation of the BWRX-300 facility (Reference 2.11-3). The information in NEDO-33914-A (Reference 2.11-4) identifies the BWRX-300 advanced civil construction and design approach.

The activities during construction and commissioning are to be monitored to identify the surfaces of civil structures that are exposed to soil, backfill or engineered fill, rock, and groundwater. The monitoring results are evaluated to determine susceptibility of the civil structures surfaces material to deterioration, and the ability to perform the intended design function under the anticipated conditions. An FIA is described in Section 4 of NEDO-33914-A (Reference 2.11-4), The FIA is further advanced specifically for the DNNP BWRX-300 in the 2023 NK054-REP-03500.8-00003 DNNP FIA report (Reference 2.11-19) by running analytical models which employed site-specific parameters that are reported in the 2022 geotechnical investigation and laboratory tests (Reference 2.11-20). The 2023 DNNP FIA report (Reference 2.11-19) analysed the subsurface soil and rock interface with the structures of the Power Block buildings including the deeply embedded RB, and new loads arising during the operational life of the BWRX-300, such as loads from ground motions, pressures, and from potential subsurface deformations that originate from subgrade instabilities and potential liquefaction (Reference 2.11-22). (Additional information on FIA as related to the DNNP and BWRX-300 is provided in Subsection 2.7.3.2, Subsection 2.7.3.3, and Subsection 2.7.5.1).

The in-service monitoring approach, presented in Section 3.3 of NEDO-33914-A (Reference 2.11-4) for the BWRX-300 also covers post-construction testing and in-service surveillance programs for below-grade structural members and foundation. Some of such activities include periodic examination of inaccessible areas, monitoring of groundwater chemistry, and monitoring of settlements and differential displacements. The purpose of the in-service monitoring programs is to monitor the condition of BWRX-300 structures over their design lives to ensure the credited safety functions as well as the overall structural integrity are maintained. The overall integrity of all civil structures, regardless of safety classification, is critical for plant personnel to safely maintain plant facilities during service and through decommissioning.

Additionally, DNNP will have a field instrumentation system related to the BWRX-300 deeply embedded RB. As described in NEDO-33914-A (Reference 2.11-4), field instrumentation that is beyond the current regulatory guidelines, is deployed to monitor the magnitude and distribution of pore pressure and amount of deformation during excavation, construction, loading and continuing through the BWRX-300 plant operation. The instrumentation provides recordings that are frequently benchmarked against design estimates. Short-term and long-term settlement monitoring plans are developed that can detect both vertical and horizontal movements in and around the structures, as well as differential distortion across the foundation footprint and differential settlements between the foundations of the CB, Turbine Building (TB), RWB and RB.

Free field and in-service seismic instrumentation are further discussed in Chapter 3, Subsection 3.3.1.5 as follows:

- Location and description of instrumentation – Subsection 3.3.1.5.1
- Design and installation – Subsection 3.3.1.5.2
- Maintenance and testing – Subsection 3.3.1.5.3
- Arrangement for control room operator notification – Subsection 3.3.1.5.4
- Comparison of measured and predicted responses – Subsection 3.3.1.5.5

#### **2.11.4 Meteorological Monitoring**

With respect to meteorological factors, data such as temperature, wind speed, and wind direction are required for monitoring the direction of dispersion of any potential containment release from the DNNP site to the surrounding environment. The meteorological data are used to calculate DRLs and dose to the public through off-site radiological environmental monitoring. In the event of an accidental release off-site, the meteorological factors provide data to support the CNEP N-PROG-RA-0001 (Reference 2.11-9).

The meteorological tower at the Darlington Nuclear site described in the 2009 NK054-REP-01210-00013 (Reference 2.11-5) is located just north of the site, just southeast of the intersection of Highway 401 and Holt Road (main access to the site). The tower has no significant obstructions from nearby buildings. Meteorological data available from the site consist of wind speed and direction at two heights (10 m and 50 m) and temperature at one height (10 m). Humidity, air pressure, and precipitation are currently not logged on-site by the meteorological tower. However, the information is readily available from Environment Canada stations as listed in Section 2.2.1 of the 2012 NK054-REP-01210-00016 (Reference 2.11-6). The data collected from the Darlington Nuclear site, per NK054-REP-01210-00013 (Reference 2.11-5) are used and adapted for to the DNNP site characteristics and the BWRX-300 design. The development of a DNNP on-site meteorological program progresses, tracked by CNSC commitment D-C-8, Meteorological Monitoring Station.

Additionally, notifications from Environment Canada for existing OPG facilities are received on severe weather which allow OPG to enter the severe weather emergency preparedness procedure N-PROC-RA-0095 (Reference 2.11-18).

#### **2.11.5 Hydrological Monitoring**

The assessment of the potential flood hazards at DNNP is described in the 2022 NK054-REP-02730-00001, Flood Hazard Assessment (Reference 2.1-21)

The BWRX-300 does include precipitation as a site-related parameter for monitoring and is assessed against the flooding hazard as part of the safety analysis as the detailed design progresses, as described in the 2020 Application to renew DNNP Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.11-2). As noted in Subsection 2.11.4, precipitation is monitored through local Environment Canada weather stations.

Groundwater flow and groundwater hydrology were assessed as a part of the 2020 NK054-CORR-00531-10533 (Reference 2.11-2), and conditions monitoring with respect to hydrology, boreholes and wells were fitted with equipment for sampling and level monitoring purposes. Sections 3.5 and 3.6 of Volume 2 of the 2022 DNNP Geotechnical Investigation (Power Block) NK054-REP-01210-00175 (Reference 2.11-21) updated the information and database on groundwater flow and hydrostratigraphic units. Annual groundwater monitoring has occurred across the DNNP site study area since the original 2009 Site Evaluation NK054-REP-01210-

00011 (Reference 2.11.3). Additional information is presented on groundwater conditions, flow, and hydro-stratigraphy in Subsection 2.5.5 and Subsection 2.7.3.2.4. Further information on the groundwater monitoring program is provided in Chapter 20, Subsection 20.11.4.

Levels in Lake Ontario are monitored by various organizations, including the Canadian Hydrographic Service, National Oceanic and Atmospheric Administration and Environment Canada as described in Section 8.2 of the 2009 NK054-REP-01210-00012 (Reference 2.11-13). The water level of Lake Ontario is controlled by the International Joint Committee— a joint group between Canada and the USA. Additional information is presented in Subsection 2.5.2.1 on how Lake Ontario water level is monitored and regulated.

The current in Lake Ontario is also monitored using the Lake Current Monitoring System as described in the 2019 NK38-OM-61100 (Reference 2.11-10) which resides in the lake approximately 1.6 km offshore of the Bowmanville Water Supply Plant, east of Darlington Nuclear site. The Lake Current Monitoring System real-time current profile measurement system is used in the event of a radiological liquid emission from Operations that takes place on the DNNP site. The Lake Current Monitoring System consists of an Acoustic Doppler Current Profiler and a Remote System Manager base station. The data acquired from Lake Current Monitoring System is also applicable to the DNNP given it is part of the Darlington Nuclear site.

#### **2.11.6 Radiation Monitoring**

Radiation Monitoring is comprised of on-site, site boundary, and off-site monitoring systems and programs. Information on radiation monitoring is available in the following subsections:

1. The environmental off-site and site boundary monitoring systems and sampling programs (Environmental Monitoring Program) – Subsection 2.9.2.1
2. The TLDs that are located around the Darlington Nuclear site perimeter as well as at off-site locations – Subsection 2.9.2.2
3. The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary – Subsection 2.9.2.3
4. Site Effluent Monitoring Program – Subsection 2.9.2.4

#### **2.11.7 Environmental Monitoring**

The Darlington Nuclear Environmental Monitoring Program identifies the contaminants and physical stressors to be monitored and conducts monitoring in the environment surrounding the site. The Environmental Monitoring Program is discussed in detail in Chapter 20, Subsection 20.11.2.

#### **2.11.8 Biological Organisms and Human Induced Hazards Monitoring**

##### **2.11.8.1 Biological Organisms**

Biological hazards specific to the DNNP site are similar to those of the 2019 DNGS NK38-REP-03611-10043 (Reference 2.11-7), given the two sites proximity.

Examples of such hazards are waterborne (e.g., fish, algae, zebra-mussel, or biofouling), large animals (e.g., herds of deer) or flying birds/insects (e.g., flocks of geese). These biological hazards are monitored and controlled in a manner enabling the safe operation of the plant.

Biofouling control typically involves appropriate biomonitoring and application of appropriate biocides/antimicrobials specific to the circuits and sensitivity of the system components. The control of the biofilms is a standard operational procedure at facilities supplied by water from Lake Ontario, and accordingly this form of biofouling is manageable for the BWRX-300 using available

technology, as described in the 2009 DNNP Site Evaluation on nuclear safety considerations NK054-REP-01210-00008 (Reference 2.11-8).

Additional information on the impact of biological and animal hazards on the safe operation of BWRX-300 facility is provided in Subsection 2.2.7.1, and on potential biofouling hazard and its impact on cooling lake water supply is presented in Subsection 2.5.2.2.

#### **2.11.8.2 Human Induced Hazards**

With respect to non-malevolent human induced hazards, all events were screened out, per the 2019 Hazards Screening Analysis NK38-REP-03611-10043 R003 (Reference 2.11-7) from the need to perform a PSA. As discussed in the following subsections, human induced hazards are screened out qualitatively or quantitatively based on the design and robustness of the BWRX-300 facility. No specific parameters are to be monitored for external human induced hazards.

##### ***2.11.8.2.1 Air Transportation Activities***

As discussed in Subsection 2.2.3.1, hazards from air transportation accidents are screened out. No site-specific parameter is expected to be monitored for aircraft/flight impacts for the DNNP site. Refer to Subsection 2.2.3.1 for additional information.

##### ***2.11.8.2.2 Chemical Explosions***

The DNNP site has various shipping lanes, which carry bulk marine shipments and the Canadian National Railway which runs within the exclusion zone of the site. The probability of accidents posing significant threat to the site is low, per the 2019 NK38-REP-03611-10043 (Reference 2.11-7). Transport vehicles carrying toxic and hazardous materials (mainly gaseous) pose a threat to worker safety which is recognized in the Site Evaluation. No site-specific parameter is expected to be monitored for chemical explosions for impacts on the DNNP site. For additional information on hazards resulting from transportation accidents refer to Subsections 2.2.3.2, 2.2.3.3, and 2.2.3.4, and from stationary non-nuclear accident refer to Subsection 2.2.4.

##### ***2.11.8.2.3 Activities at Nearby Industrial and Other Facilities***

The St. Marys Cement plant is located on the east side of DNNP site, about 700 meters from the proposed BWRX-300 location. This cement plant performs blasting at the quarry that leads to shock waves in the ground that could travel up to the BWRX-300 structures. Such shock waves are monitored using vibration monitors at a seismic monitoring station on the St. Marys property boundary. The St. Marys Cement plant is also committed to comply with the agreement established with OPG, which states that the cement plant should not carry out blasts that may exceed the maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s, per the 2019 NK38-REP-03611-10043 (Reference 2.11-7). As part of the DNGS seismic hazard curve provided in the 2021 NK38-REP-03611-10041 (Reference 2.11-1) to be used also for the DNNP site, underground shock wave effects are to be addressed through the PSA. Refer to Subsection 2.2.6 for additional information.

#### **2.11.9 Long Term Monitoring Program**

The work conducted in the 2023 report on Climate Change Impact NK054-PLAN-07007-00001 (Reference 2.11-20) confirmed the low impact of climate change stipulated in Subsection 2.5.4. Such work included climate modelling and reviewed published articles to evaluate the anticipated impact of climate change on the DNNP site and surrounding area.

Long term monitoring (periodic review/update) of applicable site-specific hazards is an inherent feature of the PSA process. As per REGDOC-2.4.2 (Reference 2.11-14), the PSA models for nuclear stations are updated every 5 years, or sooner if the facility undergoes major changes and are managed by the 2021 Preparation, Maintenance and Application of Probabilistic Safety

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Assessment N-STD-RA-033 (Reference 2.11-11). As part of this process, site-related parameters that feed into the hazard screening are revisited for new modelling methods or for any changes in the site parameters. The screening criteria for the PSA are updated every 5 years as per the 2018 OPG's Probabilistic Safety Assessment Guide N-GUID-03611-10001 (Reference 2.11-12). For cases in which data are regularly monitored at the site (e.g., wind speed or other meteorological data), and cases for which data are collected from external sources (e.g., air traffic in the vicinity of the site), the new data are assessed as part of the hazard screening for the DNGS site. A similar long-term approach is applied for the DNNP site to assess all site-related parameters for any changes.

Long term monitoring of climate change data is to be performed in accordance with REGDOC-1.1.1 (Reference 2.11-17) which requires the Site Evaluation and Site Characterization be revisited at each licensing phase to confirm it remains valid with changing environmental conditions. REGDOC-1.1.2 (Reference 2.11-15) reinforces this requirement for the Licence to Construct application and requires site characteristics be confirmed for the construction phase. REGDOC-2.5.2 (Reference 2.11-16) also requires the design of a nuclear power plant to consider all site characteristics that may affect the safety of the plant and monitoring of site-related parameters be in place throughout the lifecycle of the plant. Hazards that are applicable to the DNNP site and affected by climate change are to be monitored. Parameters associated with these climate change hazards (e.g., meteorological, lake temperature) are to be obtained from a variety of sources, including but not limited to, site-located instrumentation and local weather data. The frequency at which a climate change hazard is to be measured and analysed will depend on the nature of the hazard and its impact on the DNNP facility (e.g., nuclear safety impact, commercial impact). Climate change hazards will undergo risk assessment and where suitable will be subject to risk treatment (e.g., adaptive action or a risk monitoring plan). Where a risk monitoring plan is in place the trigger point for an adaptive action will be specified with consideration for the duration required to implement the action. The 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.11-20) provides additional information on lifecycle considerations including long term monitoring.

#### **2.11.10 References**

- 2.11-1 NK38-REP-03611-10041 R003, 2021, "Update of the OPG Darlington Site Probabilistic Seismic Hazard Assessment for the Darlington Risk Assessment (DARA)," Ontario Power Generation.
- 2.11-2 NK054-CORR-00531-10533, 2020, "Application for Renewal of OP's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation License (PRSL)," Ontario Power Generation.
- 2.11-3 NK054-REP-01210-00011 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington— Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.11-4 NEDO-33914-A, Revision 2, 2022, "BWRX-300 Advanced Civil Construction and Design Approach" GE-Hitachi Nuclear Energy Americas, LLC.
- 2.11-5 NK054-REP-01210-00013 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington— Part 4: Evaluation of Meteorological Events," Ontario Power Generation.
- 2.11-6 NK054-REP-01210-00016 R002, 2012. "Site Evaluation of the OPG New Nuclear at Darlington— Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

- 2.11-7 NK38-REP-03611-10043 R003, 2019, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 2.11-8 NK054-REP-01210-00008 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington – Nuclear Safety Considerations," Ontario Power Generation.
- 2.11-9 N-PROG-RA-0001 R019, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.11-10 NK38-OM-61100 R013, 2019, "Environmental Monitoring – Air and Water," Ontario Power Generation.
- 2.11-11 N-STD-RA-033 R006, 2021, "Preparation, Maintenance and Application of Probabilistic Safety Assessment," Ontario Power Generation.
- 2.11-12 N-GUID-03611-10001 Volume 8, 2018, "OPG Probabilistic Safety Assessment (PSA) Guide – External Hazard Screening," Ontario Power Generation.
- 2.11-13 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington-- Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.11-14 CNSC Regulatory Document REGDOC-2.4.2, "Safety Analysis Probabilistic Safety Assessment (PSA) for Reactor Facilities."
- 2.11-15 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.11-16 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.11-17 CNSC Regulatory Document REGDOC-1.1.1, "Licence Application Guide: Site Evaluation and Site Preparation for New Reactor Facilities."
- 2.11-18 N-PROC-RA-0095, "Severe Weather Emergency Preparedness," Ontario Power Generation.
- 2.11-19 NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," Ontario Power Generation
- 2.11-20 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation
- 2.11-21 NK054-REP-01210-00175 R000, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation.
- 2.11-22 NK054-REP-03500.8-00002 R000, 2022, "Darlington New Nuclear Project-- Seismically-Induced Soil Liquefaction Assessment," Ontario Power Generation

## **2.12 Ongoing Work Plans**

### **2.12.1 Introduction**

Section 2.12 details information on plans to complete ongoing DNNP specific works involving geotechnical investigations, laboratory tests, analyses, and assessments to validate and update existing DNNP parameters or generate new site-specific characterizations and parameters to supplement and update existing database. Each disposition plan provides:

- Background information on the ongoing work
- The schedule and workflow by which the ongoing work is to be completed
- Risks associated with the ongoing work
- Chapter 2 sections impacted by the ongoing work
- Progress of work, including deliverables

Details of each work is provided as follows:

Subsection 2.12.2 – Foundation Interface Analysis (FIA)

Subsection 2.12.3— Site Geotechnical and Seismic Hazard Investigation Plan, which includes

- Geotechnical investigations (Power Block) and laboratory tests
- Offshore geotechnical investigation
- Site-specific Probabilistic Seismic Hazard Assessment (PSHA)
- Seismically-induced liquefaction assessment

Subsection 2.12.4— Flood Hazard Assessment

Subsection 2.12.5 – Climate Change Impact

Subsection 2.12.6 – 3-second Wind Gust Validation

Subsection 2.12.7- Winter PMP Validation

Subsection 2.12.8 – PMP Validation

The results of each completed work are incorporated into the impacted sections in Chapter 2. A summary description of each work along with the deliverables are provided in Table 2.12-1.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.2	<p>Foundation Interface Analysis (FIA)</p> <p><u>Status:</u> Complete</p>	<p>The FIA results will support the evaluation of the construction plan, the stability of the excavation, ground improvements and the design of excavation support systems. Also, the results of ground pressure demands on the below-grade exterior walls of the RB will be used to validate ground pressure design loads. The FIA will be performed with three dimensional models representing the site conditions at all project stages, including design, construction, and operation.</p> <p>Specific tasks are as follows:</p> <ul style="list-style-type: none"> <li>• Evaluation of the subgrade materials and the materials surrounding the deeply embedded BRWX-300 RB</li> <li>• Confirmation that the Radwaste Building, Turbine Building, and CB foundations are to be supported by the engineered fill, intermediate glaciolacustrine, and lower till soils</li> <li>• Confirmation of the stability of sand and rock excavation for the stability of the deeply embedded RB shaft evaluation for excavation and construction</li> </ul> <p>The resulting report will discuss:</p> <ol style="list-style-type: none"> <li>1. Effects of excavation, dewatering (based on hydrogeology report) and construction on subgrade material properties</li> <li>2. Evaluations of potential for unstable rock mass or unstable blocks and wedges including the joints and sizes of the potential blocks or wedges</li> <li>3. Results of the FIA of the site characterization, excavation, construction, loading, operation stages</li> <li>4. Inputs and results of sensitivity FIA or additional stability analysis</li> </ol> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interaction Analysis (FIA) Report," Ontario Power Generation</li> </ol>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.3	Site Geotechnical and Seismic Hazard Investigation Plan  <u>Status:</u> Complete	<p>The main deliverables of OPG's Site Geotechnical and Seismic Hazard Investigation are as follows:</p> <ol style="list-style-type: none"> <li>1. Perform Geophysical Survey and Mapping of Subsurface Strata</li> <li>2. Detailed Site Investigation and Geotechnical Lab Tests</li> <li>3. Excavation and Stockpile / Earth Removal</li> <li>4. Geological Hazard Scenarios</li> <li>5. Liquefaction Potential Assessment</li> <li>6. DNNP Probabilistic Seismic Hazard Analysis</li> <li>7. DNNP Specific Seismic Hazard</li> </ol> <p>The results of this work will be used for the confirmation of BWRX-300 bounding parameters</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. (NK054-REP-01210-00175) Golder Associates Ltd. (Golder), 2022, Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project, Revision 2, Volumes 1 and 2, July 29</li> <li>2. (NK054-REP-10180-00001) Golder Associates Ltd. (Golder), 2023, Offshore Geotechnical Investigation Darlington New Nuclear Project, Revision 0.</li> <li>3. (NK054-REP-03500.8-00001) Kinectrics Inc., K-620423/RP/0001 R01, "Darlington New Nuclear Project-- Site-Specific Probabilistic Seismic Hazard Assessment," 2022</li> <li>4. (NK054-REP-03500.8-00002) Kinectrics Inc., K-620423/RP/0002 R00, "Darlington New Nuclear Project-- Seismically-Induced Soil Liquefaction Assessment," 2022</li> </ol>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.4	Flood Hazard Assessment <u>Status:</u> Complete	<p>This Hydrological Analysis is expected to follow a similar format to the original flood assessment covering:</p> <ul style="list-style-type: none"> <li>• Identification of Flooding Hazards</li> <li>• Description of DNNP Site Layout</li> <li>• Assessment of Flooding Hazards</li> <li>• Flood Protection</li> <li>• Modification of the Flood Hazard over time</li> <li>• Monitoring and Warning for Plant Protection</li> <li>• Conclusions and Recommendations</li> </ul> <p>The results of this work are used for the confirmation of BWRX-300 bounding parameters</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. NK054-REP-02730-00001 R000, 2022, "Flood Hazard Assessment", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-001 R01).</li> </ol>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.5	<p>Climate Change Impact</p> <p><u>Status:</u> “CNSC Deliverable1: DNNP Strategy for Addressing Climate Change Impacts.”</p> <p>Complete</p>	<p>Conditions from climate change which impact flooding have been incorporated into Chapter 2 based on the 2022 NK054-REP-02730-00001 “Flood Hazard Assessment” (Reference 2.12-4).</p> <p>OPG has issued the 2023 NK054-PLAN-07007-00001 R000 “Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts” (Reference 2.12-5). This strategy has two primary phases: Climate Change Risk Assessment and Climate Change Risk Treatment. Following work will be performed on an as-required basis to integrate climate change assessments into the current nuclear safety framework. This will include lifecycle considerations such as long-term monitoring and periodic reassessment of hazards associated with climate change DNNP commitment D-C-7 in accordance with the strategy outlined in NK054-PLAN-07007-00001 (Reference 2.12-5). D-C-7 will be completed prior to start of construction as per NK054-REP-01210-00078 (Reference 2.12-2).</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. (NK054-PLAN-07007-00001 R000), 2023, “Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts”, Ontario Power Generation</li> <li>2. NK054-REP-07007-1049426 R001, 2023, “Darlington New Nuclear Project – Hazard Bounding Analysis,” Ontario Power Generation</li> <li>3. NK054-REP-07007-1028871 R000, 2022, “Darlington New Nuclear Project— Gradual Climate Change and Natural Hazard Identification,” Ontario Power Generation</li> </ol>
2.12.6	<p>3-second Wind Gust Calculation</p> <p><u>Status:</u> Complete</p>	<p>While maximum wind speed is an instantaneous wind speed, the 3-second gust value is a sustained wind speed. Maximum wind speed is shown in Subsection 2.6.5.</p> <p>Key two aspects of this work are:</p> <ul style="list-style-type: none"> <li>• Calculation of the site characteristic for 3-second wind gust speed is in progress</li> <li>• Value will confirm BWRX-300 bounding approach</li> </ul> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. NK054-REP-02730-00003 R000, 2022, “Wind Gust Analysis”, Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-003 R01)</li> </ol>

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.7	Winter PMP Validation  <u>Status:</u> Complete	Work started to finalize appropriate consideration for snow load with a Winter Probable Maximum Precipitation (PMP) event. DNNP considers this a review level condition.  Finalization of the coincident snow load and winter PMP is complete.  <u>Deliverables:</u>  1. NK054-REP-02730-00004 R000, 2022, "Winter PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-004 R01)
2.12.8	PMP Validation  <u>Status:</u> Complete	Confirmation of rainfall and PMP  <u>Deliverables:</u>  1. NK054-REP-02730-00002 R000, 2022, "PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-002 R01)

## **2.12.2 Foundation Interface Analysis**

### **2.12.2.1 Background**

OPG has undertaken a site-specific, non-linear FIA, to ensure the stability of structures, supporting media, soil, and rock per NUREG-800 SRP 2.5.4 guidance. The FIA results support the evaluation of the construction plan, the stability of the excavation, ground improvements and the design of excavation support systems. Also, the results of ground pressure demands on the below-grade exterior walls of the RB are used to validate ground pressure design loads. The FIA is performed with three dimensional models representing the site conditions at all project stages, including design, construction, and operation.

The schematic workplan for the FIA modelling is shown in Figure 2.12.2-1.

All relevant available reports describing ground conditions and structural details are reviewed including but not limited to: Geotechnical Investigation Factual and Interpretation Reports, NEDO 33914 Licensing Topical Report [1], and relevant nuclear standards/guidelines. The factual data are summarized and classified for each geological unit and the input parameters required for FIA numerical modelling are calculated or extracted from the laboratory and in-situ test results. The structural information such as the shoring design, construction staging, and the structure details are reviewed and summarized in our FIA interaction modelling activity.

All relevant available reports describing ground conditions and structural details are reviewed, including but not limited to:

- Geotechnical Investigation Factual and Interpretation Reports, NEDO 33914 Licensing Topical Report (Reference 2.12-1)
- Relevant nuclear standards/guidelines.

This information is used to develop the Finite Element Analysis method and 3D framework in Plaxis 3D, allowing full FIA interaction modelling.

The Technical Report is prepared based on the FIA modelling, includes the results of the FIA of the deeply embedded BWRX-300 RB and the surrounding Power Block foundations at the DNNP site. The report discusses:

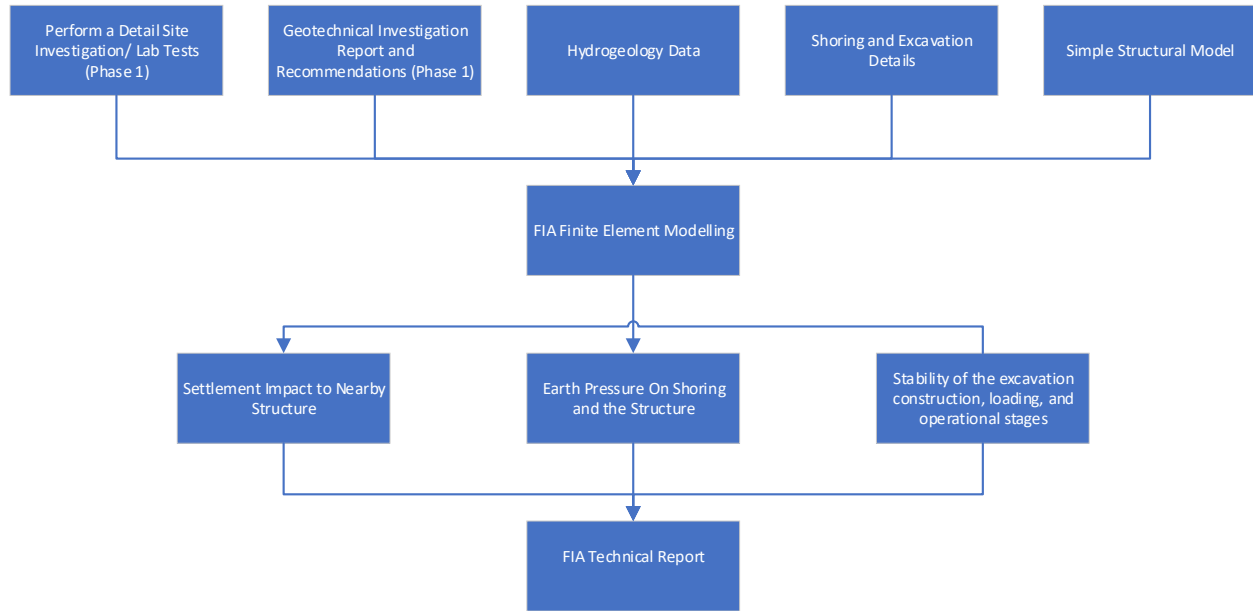
1. Effects of excavation, dewatering (based on hydrogeology report) and construction on subgrade material properties
2. Evaluations of potential for unstable rock mass or unstable blocks and wedges including the joints and sizes of the potential blocks or wedges
3. Results of the FIA of the site characterization, excavation, construction, loading, operation stages
4. Inputs and results of sensitivity FIA or additional stability analysis

### **2.12.2.2 Project Schedule and Logic**

The report concludes the results of the FIA for the deeply embedded BWRX-300 RB and the surrounding Power Block foundations at the DNNP site. The schematic workplan for the FIA modelling is shown in Figure 2.12.2-1.



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 2.12.2-1: FIA Modelling Workflow and Deliverables**

#### **2.12.2.3 Risks**

Project timeline is dependent on DNNP confirmatory geotechnical investigation results (Laboratory Test Results and In-Site Test Results) (refer to Subsection 2.12.3). Any delays to the geotechnical investigation may cause a delay to the FIA final deliverable (Technical Report)

#### **2.12.2.4 Impacted Chapter 2 Sections**

Section 2.7 – Geology, Seismology, and Geotechnical Engineering.

#### **2.12.2.5 Progress of Work**

1. Review completed of recent reports by Golder Associates Ltd. (refer to Subsection 2.12.3) that includes site-specific results of geotechnical investigations and laboratory tests
2. Information received on shoring and excavation details from AECON
3. A simple structural model is tested and verified
4. FIA Finite Element modelling is developed
5. Technical memoranda developed, circulated for review and comments, on the following topics:
  - a. Bearing Capacity Evaluations of the BWRX-300 RB and the Surrounding Power Block Foundations at the DNNP Site
  - b. Settlement Evaluations of the BWRX-300 RB and the Surrounding Power Block Foundations at the DNNP Site
  - c. Excavation and Construction Stages of the BWRX-300 RB Shaft
  - d. FIA Numerical Modelling
6. Additional key parameters are sought and confirmed for use as input to the FIA model

7. Final report is complete

Work is complete and closed. The results are incorporated in Section 2.7.

Deliverables:

The following report was submitted by the outsourced contactor, and was reviewed and accepted by to OPG:

1. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," Ontario Power Generation

### **2.12.3 Site Geotechnical and Seismic Hazard Investigation Plan**

#### **2.12.3.1 Background**

##### Geotechnical Program

OPG has undertaken a detailed site geotechnical program which provides information on the soil physical, mechanical, and dynamic properties of overburden and rock material. The program assesses whether there are karstic features in the local bedrock at the site. The program is linked to the existing CNSC commitment D-P-9 Site Geotechnical and Seismic Hazard Investigation (Reference 2.12-2). The schematic workplan for OPG's Geotechnical Program is shown in Figure 2.12.3-1.

The geotechnical and seismic hazard investigation program, undertaken by OPG, has primary goals to gather sufficient geological data for the proposed DNNP site, identify potential geotechnical and seismic related hazards, and perform the necessary safety evaluations, analyses, and assessments. Investigation methods used included compilation, review and evaluation of existing/historical documents, detailed geophysical and geotechnical site exploration, and extensive in-situ and laboratory testing. Each of these methods are applicable to all stages of the Site Evaluation process, but to varying extents. The main deliverables of OPG's Site Geotechnical and Seismic Hazard Investigation are as follows:

- Perform Geophysical Survey and Mapping of Subsurface Strata
- Detailed Site Investigation and Geotechnical Lab Tests
- Excavation and Stockpile / Earth Removal
- Geological Hazard Scenarios
- Liquefaction Potential Assessment
- DNNP Probabilistic Seismic Hazard Analysis
- DNNP Specific Seismic Hazard

The results of the OPG's Geotechnical and Seismic Hazard Investigation feed into Section 2.7 Geology, Seismology, and Geotechnical Engineering.

#### **2.12.3.2 Project Schedule and Logic**

OPG's Geotechnical Program for Phase 1 is demonstrated in the Project Logic of Figure 2.12.3-1. DNNP's Geotechnical and Seismic Investigations are linked to the existing DNNP CNSC commitment D-P-9 Site Geotechnical and Seismic Hazards Investigations (Reference 13-3).

#### **2.12.3.3 Risks**

Delays in completing this program may impact completing OPG work on FIA discussed in Subsection 2.12.2.

#### **2.12.3.4 Impacted Chapter 2 Sections**

Subsection 2.7.3 Geotechnical Characteristics

#### **2.12.3.5 Progress of Work**

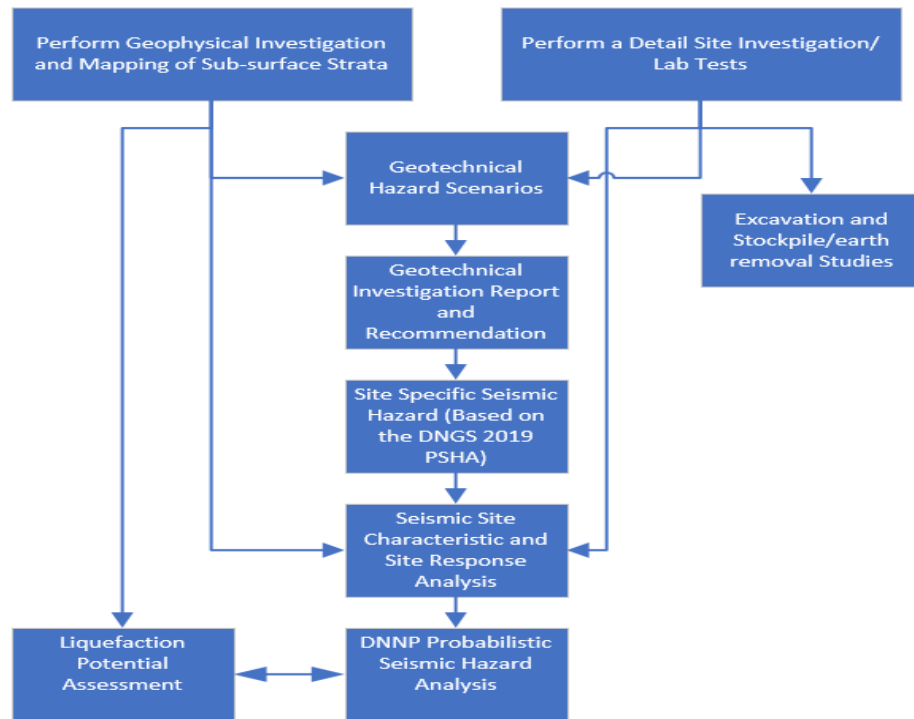
1. Completed geophysical investigation and mapping of subsurface strata
2. Completed detailed site investigation and laboratory tests
3. Drafted report on the geophysical investigation and laboratory tests as well as recommendations
4. Excavation and earth removal studies continue
5. Site-specific characteristics and site response analysis is progressing
6. DNNP PSHA is progressing
7. Liquefaction potential is being assessed and is progressing

Work is complete and closed. The results are incorporated in Section 2.7.

#### **Deliverables**

The reports were submitted by the outsourced contactor, and were reviewed and accepted by OPG:

1. NK054-REP-01210-00175 R01, (Golder 2022) "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project," Volumes 1 and 2, Ontario Power Generation.
2. (NK054-REP-10180-00001) Golder Associates Ltd. (Golder), 2023, Offshore Geotechnical Investigation Darlington New Nuclear Project, Revision 0.
3. NK054-REP-03500.8-00001 R00, 2022, Kinectrics Inc., K-620423/RP/0001 R01, "Darlington New Nuclear Project— Site-Specific Probabilistic Seismic Hazard Assessment," Ontario Power Generation.
4. (NK054-REP-03500.8-00002) Kinectrics Inc., K-620423/RP/0002 R00, "Darlington New Nuclear Project— Seismically-Induced Soil Liquefaction Assessment," 2022



**Figure 2.12.3-1: Workflow for the Geotechnical Program**

## **2.12.4 Flood Hazard Assessment**

### **2.12.4.1 Background**

A Flood Hazard Assessment is required for Section 2.5 Hydrology.

A previous DNNP Flood Hazard Assessment was completed (Reference 2.12-4) as part of the original Site Evaluation in 2009 included in the EIS and Licence to Prepare Site process, which reflects a site build for up to 4800 mWe of either an EPR, AP-1000, ACR or EC-6 reactor type.

The construction of a 300 mWe BWRX-300 Small Modular Reactor at the DNNP site, led to different site layout, plant grade, and topography to that previously evaluated in Reference 2.12.4. This requires an update to the Flood Hazard Assessment.

OPG has contracted an outsource to complete the Hydrological Analysis which followed a similar format to the original flood assessment covering:

- Review of existing work and data
- Completion of a gap analysis to determine if additional modelling and analysis is required
- Completion of required modelling and analysis
- Organization of information, identification of flood hazards and mitigations, meeting the requirements outlined in REG-DOC1.1.1 and IAEA Nos. NS-R-3, SSG-18, and other regulatory documents

- Identification of Flooding Hazards
- Description of DNNP Site Layout
- Assessment of Flooding Hazards
- Flood Protection
- Modification of the Flood Hazard over time
- Monitoring and Warning for Plant Protection
- Conclusions and Recommendations

#### **2.12.4.2 Project Schedule and Logic**

The following deliverables close this ongoing work:

- Draft Flood Hazard Assessment report
- Final Flood Hazard Assessment report

#### **2.12.4.3 Risks**

None.

#### **2.12.4.4 Impacted Chapter 2 Sections**

Section 2.5 Hydrology.

#### **2.12.4.5 Progress of Work**

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00001, "Flood Hazard Assessment," Ontario Power Generation.
3. The report has the following contents
  1. Introduction
  2. General Site Description and Characteristics
  3. Existing Site Conditions – Potential Flood Hazards
  4. Post-Development Site Layout
  5. Assessment of Flood Hazards
  6. Mitigation Measures
  7. Modification of the Flood Hazard with Time
  8. Monitoring and Warning for Plant Protection
  9. Conclusions and Recommendations
  10. References

Work is complete and the results are incorporated in impacted sections of Chapter 2

#### ***Deliverables:***

1. NK054-REP-02730-00001 R000, 2022, "Flood Hazard Assessment", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-001 R01).

## **2.12.5 Climate Change Impact**

### **2.12.5.1 Background**

The potential effects of climate change on external natural hazards such as flooding and temperature as well as life cycle considerations including long-term monitoring programs (refer to Subsection 2.5.4, Subsection 2.6.4, Subsection 2.6.12, and Subsection 2.11.9) are linked to the existing commitment D-C-7, Contingency Plan for Flooding and Other Extreme Weather Hazards (Reference 2.12-2). To address this commitment, OPG has developed NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.12-5), which describes the plan for fulfilling the requirements of commitment D-C-7, and consequently ensuring the DNNP facility is resilient to climate change hazards. Additional information on long term monitoring of climate change hazards is provided in Section 2.11.9.

The DNNP Strategy for addressing Climate Change Impact consists of the following three phases:

1. Phase 1 – Climate Change Risk Assessment

The purpose of this phase is to perform a climate change risk assessment for the DNNP facility to identify climate change hazards, bounding values/ranges, and vulnerable structures, systems, and components. There are two main activities in this phase, the Hazards Identification and Bounding Analysis. Hazard Identification will identify climate change related hazards that can affect DNNP site (e.g., hydrological, meteorological, etc.). Bounding Analysis report will then determine bounding values/ranges for the hazards that pose nuclear safety, commercial, or operational impacts. The values from the bounding analysis will feed into the Plant Envelope Assessment to determine which systems may be vulnerable to climate change hazards.

2. Phase 2 – Climate Change Risk Treatment

The phase analyses the design margins of vulnerable structures, systems and components and develops risk treatments as required. These risk treatments can include adaptation of the design or implementation of risk monitoring plans. The completion of Phase 2 will provide the necessary information that will comply with addressing the effects of climate change on-site.

3. Phase 3 – As Required Work

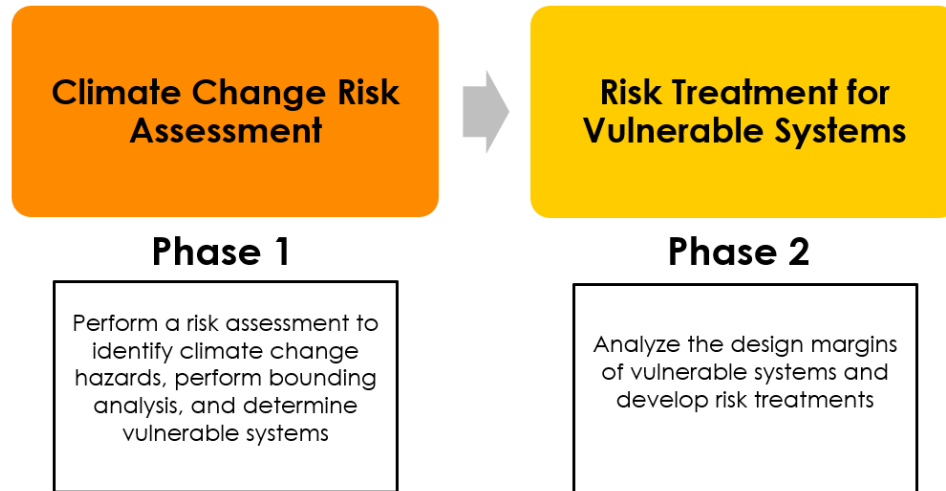
Work will be performed on an as required basis to integrate climate change assessments into the current nuclear safety framework.

The results of this work are used to confirm low impact of climate change. Where structures, systems, and components are potentially vulnerable to climate change hazards, appropriate risk treatments are developed to ensure climate change resilience is implemented within the design.

To ensure alignment with the regulator, OPG will submit three deliverables to the CNSC. The first being the 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.12-5) which provides the CNSC a description of the proposed methodology for the close-out of commitment D-C-7. The second deliverable will be a summary report of Phase 1, which outlines the results from the Hazard Identification, Bounding Analysis, and Plant Envelope Assessment. The Phase 1 report will be submitted to the CNSC to progress closure of D-C-7. Lastly, the third deliverable will be a summary report of Phase 2, which will summarize the risk assessment of vulnerable structures, systems, and components and their risk treatment plans. The Phase 2 report will be submitted to the CNSC for closure of DNNP commitment D-C-7. CNSC feedback will be obtained on strategy and deliverables for D-C-7 prior to licence to the start of construction.

### 2.12.5.2 Project Logic

Phase 1 Climate Change Risk Assessment and Phase 2 Risk Treatment for Vulnerable Systems are to be completed in 2023. This work will be tracked according to the 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts to align with the closing of existing commitment D-C-7 prior to the start of construction.



**Figure 2.12.5-1: Risk Roadmap for OPG Strategy on Addressing Climate Change Impacts**

### 2.12.5.3 Risks

None.

### 2.12.5.4 Impacted Chapter 2 Sections

Subsection 2.6.2 Temperature

Subsection 2.6.4 Rainfall

Subsection 2.11.9 Long Term Monitoring Program

### 2.12.5.5 Progress of Work

OPG issued, in January 2023, the plan as NK054-PLAN-07007-0001, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation

The plan has the following contents

1. Introduction
2. Objective
3. Regulatory and Governance Drivers
4. Strategy Overview
5. Lifecycle Considerations
6. Strategy Partners
7. Definitions and Acronyms
8. References

Work is complete and results are incorporated in Subsection 2.5.4, Subsection 2.6.4, Subsection 2.6.12, and Subsection 2.11.9

Phase 1 and 2 of the 2023 NK054-PLAN-07007-0001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts are to be completed and tracked to the existing commitment D-C-7.

**Deliverables:**

1. (NK054-PLAN-07007-00001 R000), 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts", Ontario Power Generation.
2. NK054-REP-07007-1049426 R001, 2023, "Darlington New Nuclear Project – Hazard Bounding Analysis," Ontario Power Generation
3. NK054-REP-07007-1028871 R000, 2022, "Darlington New Nuclear Project— Gradual Climate Change and Natural Hazard Identification," Ontario Power Generation

## **2.12.6 3-Second Wind Gust Speed**

### **2.12.6.1 Background**

Chapter 2, Subsection 2.6.5 requires description of the site characteristic for 3-second wind gust speed. While maximum wind speed is an instantaneous wind speed, the 3-second gust value is a sustained wind speed. Maximum wind speed is shown in Subsection 2.6.5.

Calculation of the site characteristic for 3-second gust wind is in progress and will be added in a future revision.

### **2.12.6.2 Project Logic**

Completion of calculations is undergoing and will be updated in the subsequent revision of PSAR Chapter 2.

### **2.12.6.3 Assumptions**

None.

### **2.12.6.4 Risks**

None.

### **2.12.6.5 Impacted Chapter 2 Sections**

Subsection 2.6.5 – Wind Speed

### **2.12.6.6 Progress of Work**

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00003, "Wind Gust Analysis," Ontario Power Generation.
3. The report has the following contents
  1. Introduction
  2. Study Site and Data
  3. Wind Rose Diagram
  4. Frequency Analysis



5. Conclusions
6. References

Work is complete and results are incorporated in impacted sections Chapter 2

Deliverables:

1. NK054-REP-02730-00003 R000, 2022, "Wind Gust Analysis", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-003 R01)

## **2.12.7 Snow Load and Coincident Winter Probable Maximum Precipitation**

### **2.12.7.1 Background**

Work is ongoing to finalize appropriate consideration for snow load with a Winter PMP event. DNNP considers this a review level condition.

### **2.12.7.2 Project Logic**

Completion of calculations is undergoing and will be updated in a subsequent revision of PSAR Chapter 2.

Winter PMP Validation - The requirements of N291 for safety related structures other than containment for 100 years snow loading is not mentioned nor the guidance in it to extrapolate the National Building Code of Canada (NBCC) 50-years value if the 100-years site snow values are not available. N291 mention this for the snow component, however, it is silent about associated rain.

For safety related structures, 100 years snow with 100 years associated rain would be required for the design.

It is recommended that OPG follow the General-Electric Hitachi recommendation in the Design Input Request for Non-Seismic External Hazards at DNNP Site document to determine the following site-specific parameters:

- 100-year return period ground snowpack
- Historical maximum snowpack, including the month of occurrence • 100-year return period ground snowfall
- Historical maximum ground snowfall
- 48-hour Winter PMP over a 25.9-square-kilometer (10-square-mile) area at this location during those months with the historically highest snowpacks.

The depth, area, and duration curves of the probable maximum storm event equivalent to the Winter PMP should be identified. (OPG, 2017)

The anticipated resulting roof loading will be situated in the range of 3.0-4.5 kPa.

### **2.12.7.3 Risks**

None.

### **2.12.7.4 Impacted Chapter 2 Sections**

Subsection 2.6.9 – Snow and Ice Load

### **2.12.7.5 Progress of Work**

1. Work is completed and a final report is delivered and accepted by OPG

2. OPG issued, in December 2022, the report as NK054-REP-02730-00004, "Winter PMP Validation," Ontario Power Generation.
3. The report has the following contents
  1. Introduction
  2. Existing Values
  3. Winter PMP Usage
  4. Conclusions
  5. References

Work is complete and results are incorporated in impacted sections Chapter 2

**Deliverables:**

1. NK054-REP-02730-00004 R000, 2022, "Winter PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-004 R01)

**2.12.8 Confirmation of Probable Maximum Precipitation**

**2.12.8.1 Background**

Subsection 2.6.4 describes the rainfall and PMP for the Darlington Nuclear site (which includes the DNNP site). Also, Subsection 2.12.4 describes an ongoing work to update the PMP and Probable Maximum Flood for the DNNP site for BWRX-300 unit 1, with potential three additional units.

This information is being supplemented by PMP Validation work being added to Subsection 2.12.4. The supplementary work is to satisfy the requirements of N291 of 100 years return period for safety related structures (similar to wind and snow), and to ensure information in: the recommendation of 21 mm for storm H (in Table 3-1 of the contractor's preliminary report) meets the NBCC as a minimum (as NBCC value for 15 min is 23mm).

**2.12.8.2 Project Schedule and Logic**

Confirmation work is ongoing. Subsection 2.6.4 is expected to be updated, as required, in the subsequent revision of the PSAR Chapter 2.

**2.12.8.3 Assumptions**

None

**2.12.8.4 Risks**

None

**2.12.8.5 Impacted Chapter 2 Sections**

Subsection 2.6.4 – Rainfall

**2.12.8.6 Progress of Work**

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00002, "PMP Validation," Ontario Power Generation.
3. The report has the following contents
  1. Introduction

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

2. Storms
  - a. PMP Validation
  - b. Plant Parameter Envelop Storms
  - c. National Building Code of Canada Storms
3. Conclusions
4. References

Work is complete and results are incorporated in impacted sections Chapter 2

**Deliverables:**

1. NK054-REP-02730-00002 R000, 2022, "PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-002 R01)

**2.12.9 References**

- 2.12-1 NEDO-33914-A, Revision 2, 2022, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC.
- 2.12-2 NK054-REP-01210-00078 R007, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 2.12-3 NK054-PLAN-01210-00033, Site Geotechnical and Seismic Hazard Investigation Plan," Ontario Power Generation.
- 2.12-4 NK054-REP-01210-00012-R01, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.12-5 NK054-PLAN-07007-00001 R000, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts", Ontario Power Generation.

## **2.13 Appendices**

Appendix A List of Industrial Facilities within the Survey Area

Appendix B List of Roads within the Survey Area

Appendix C List of Park Spaces and Water Bodies within the Survey Area

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**APPENDIX A – List of Industrial Facilities within the Survey Area**

<b>Company Name</b>	<b>Location</b>
McAshpalt Industries Ltd. - Oshawa Facility	Bottom of Farewell Street
Gerdau Metals Recycling - Oshawa	Waterloo Crt
TMT Salvage & Metal Recyclers	SE Corner - Nelson St & Waterloo Crt
D. Crupi & Sons Ltd.	NE Corner - Nelson St & Wellington Ave E.
Allmix Concrete Oshawa	NE Corner - Farewell St & Harbour Rd.
Coco Paving Plant	SE Corner - Wilson Rd N & Taunton Rd
Covanta Durham York	Courtice Rd. & Megawatt Dr
Courtice Water Pollution Control Plant (WPCP)	Osbourne Rd.
Miller Compost	Baseline Rd & Hancock Rd.
Hydro One Bowmanville SS	Toward bottom of Holt Rd.
St. Marys Cement Group	Bottom of Bowmanville Ave.
CBM Aggregates	Waverley Rd.
Port Darlington WPCP	E Shore Dr.
Bowmanville Water Supply Plant	E Beach Rd.

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**APPENDIX B – List of Roads within the Survey Area**

<b>Name of Road / Highway / Station</b>	<b>Direction</b>	<b>Road Type</b>
Highway 401	W-E	Hwy
Highway 418	N-S	Hwy
Highway 407	W-E	Hwy
Baseline Road W	W-E	Arterial
Courtice Road	N-S	Arterial
2nd Line W	Internal	Minor Arterial
Park Rd	N-S	Minor Arterial
Energy Dr	W-E	Arterial
Symons Rd	N-S	Minor Arterial
Crago Rd	N-S	Minor Arterial
Megawatt Dr	W-E	Minor Arterial
Osbourne Rd	N-S	Minor Arterial
Darlington Park Rd	W-E	Minor Arterial
Down Rd	N-S	Minor Arterial
Holt Rd	N-S	Arterial
Martin Rd S	N-S	Minor Arterial
Colonel Sam Dr	W-E	Minor Arterial
Cedar Crest Beach Rd	W-E	Minor Arterial
Cove Rd	W-E	Minor Arterial
W Beach Rd	N-S	Minor Arterial
Main St	W-E	Minor Arterial

<b>Name of Road / Highway / Station</b>	<b>Direction</b>	<b>Road Type</b>
E Beach Rd	W-E	Minor Arterial
Port Darlington Rd	N-S	Minor Arterial
Lake Rd	W-E	Minor Arterial
S Service Rd	N-S	Minor Arterial
Lookout Dr	W-E	Minor Arterial
Bennett Rd	N-S	Arterial
Wilmot Creek Dr	N-S	Minor Arterial
Heatherlea Dr		Residential
Hinkley Tr		Residential
Cliff Dr		Residential
Fir Dr		Residential
Niagara Tr		Residential
Wilmot Tr	W-E	Minor Arterial
Little Brook Rd		Residential
Bluffs Rd		Residential
Heritage Ln		Residential
The Cove Rd		Residential
Steelhead Ln		Residential
Fairway Dr		Residential
Service Rd	W-E	Minor Arterial
Bloor St E	W-E	Arterial
Farewell St	N-S	Minor Arterial
Veterans Rd	W-E	Minor Arterial
Wilson Rd S	N-S	Minor Arterial
Raleigh Ave	W-E	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Wentworth St W	W-E	Arterial
Marwood Dr	W-E	Minor Arterial
Harbour Rd	W-E	Arterial
Drake St	N-S	Minor Arterial
Holland St	N-S	Minor Arterial
Simcoe St S	N-S	Arterial
Nelson St	N-S	Minor Arterial
Ritson Rd S	N-S	Arterial
Dnipro Blvd	W-E	Minor Arterial
Conant St	W-E	Residential
Sylvia St		Residential
Myers St		Residential
Sharon Ave		Residential
Trafalgar Ave		Residential
Waterloo St / Crt		Residential
Tilbury St		Residential
Wellington Ave E		Residential
Kawartha Ave	W-E	Minor Arterial
Southlawn Ave	W-E	Minor Arterial
Cloverdale St	N-S	Minor Arterial
Grassmere Crt		Residential
Ravine Rd	N-S	Minor Arterial
Sandra St W/E		Residential
Wolfe St	W-E	Minor Arterial
Daniel St		Residential
Douglas St	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
4th Ave	W-E	Minor Arterial
Annis St	W-E	Minor Arterial
Rowena St	N-S	Minor Arterial
Gifford St		Residential
Phillips St		Residential
Merritt St		Residential
Knights Rd		Residential
Cedar St		Residential
Erie St		Residential
Whiting Ave	N-S	Minor Arterial
Robson St		Residential
Frank St		Residential
Valley Dr	W-E	Minor Arterial
Wecker Dr	W-E	Minor Arterial
Outlet Dr		Residential
Birchcliffe Ave	N-S	Minor Arterial
Kluane Ave	N-S	Minor Arterial
Rondeau Crt		Residential
Madawaska Ave		Residential
Sauble St		Residential
Quetico Ave / Crt		Residential
Georgian Crt		Residential
Fundy St / Crt		Residential
Phillip Murray Ave		Residential
Chaleur Ave		Residential
Sharbot St		Residential
Minden St		Residential
Scugog Ave		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Jasper Ave		Residential
Banff Ave		Residential
Geneva Ave		Residential
Thomas St	W-E	Minor Arterial
Tamarack Crt		Residential
Erie St		Residential
Grandview Dr	W-E	Minor Arterial
Downview Cres	W-E	Minor Arterial
Endna Crt		Residential
Welsey Dr	N-S	Minor Arterial
Down Cres		Residential
Norman Cres		Residential
Southdown Dr	N-S	Minor Arterial
Southdale Ave	W-E	Minor Arterial
Southgate Dr	N-S	Minor Arterial
Southridge St	N-S	Minor Arterial
Southport Dr		Residential
Townline Rd S	N-S	Minor Arterial
Gord Vinson Ave	W-E	Minor Arterial
Kilgannon Ave		Residential
Pickard Gate		Residential
Cornish Dr	N-S	Minor Arterial
Staples Ave		Residential
Bingham Gate		Residential
Dudley Crt		Residential
Cousins St		Residential

Name of Road / Highway / Station	Direction	Road Type
Fenning Dr	N-S	Minor Arterial
Stainton St		Residential
Roy Nichols Dr		Residential
Southfield Ave		Residential
Aylesworth Ave		Residential
Montague Ave		Residential
Frank Wheeler Ave		Residential
Eastfield Cres		Residential
Rosswell Dr		Residential
Dewell Cres		Residential
Bathgate Cres		Residential
Kersey Cres		Residential
Prestonvale Rd	N-S	Arterial
Trulls Rd	N-S	Arterial
Cigas Rd	W-E	Minor Arterial
Hancock Rd	N-S	Arterial
McKnight Rd		Residential
Courtice Crt		Residential
Solina Rd	N-S	Arterial
Rundle Rd	N-S	Arterial
Maple Grove Rd	N-S	Arterial
Boswell Dr	N-S	Minor Arterial
Ivory Crt		Residential
Shady Lane Cres		Residential
Bonathan Cres		Residential
Connors Crt		Residential
Rustwood St		Residential
Weldick Cres		Residential
Padfield Dr		Residential
Hammond St		Residential



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Oxley Crt		Residential
Collier Ln		Residential
Dystra Ln		Residential
Sidney Ln		Residential
Connell Ln		Residential
Farmstead Dr		Residential
Autumn harvest Rd		Residential
McBride Ave		Residential
Buxton Ln		Residential
Buttonschaw St	N-S	Minor Arterial
Woolacott Ln		Residential
McPhail Ave		Residential
Shackleton St		Residential
Kimble Ave		Residential
Remmington St	W-E	Minor Arterial
Butson Cres		Residential
Green Rd	N-S	Arterial
Clarington Blvd	N-S	Minor Arterial
Prince William Blvd	W-E	Minor Arterial
Pethick St	N-S	Minor Arterial
Aspen Springs Dr	W-E	Minor Arterial
Baxter St	N-S	Minor Arterial
West Side Dr		Residential
Landerville Ln		Residential
Fry Cres		Residential
Vail Meadows Cres		Residential
Glen Ray Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Hartwell Ave		Residential
Candler Crt		Residential
Prestonway Dr		Residential
Bonnycastle Dr		Residential
Luttrell St		Residential
Higgon St		Residential
Brodie Crt		Residential
Martin Rd	N-S	Minor Arterial
Bagnell Cres		Residential
Abernethy Cres		Residential
Penfound Dr		Residential
Alonna St		Residential
Clancy Ln		Residential
Bottrell St		Residential
Squires Gt		Residential
Roser Cres		Residential
Walbridge Crt		Residential
Woolner Ln		Residential
Dodds Sq		Residential
Millburn Dr		Residential
Bannister St		Residential
Spicer Sq	W-E	Minor Arterial
Bowmanville Ave	N-S	Arterial
Kings Hill Ln		Residential
McCrimmon Cres		Residential
Wrenn Blvd		Residential
Rhonda Blvd	N-S	Minor Arterial
Chapel St		Residential
Roenigk Dr	W-E	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Waverley Rd	N-S	Minor Arterial
Strike Ave		Residential
Little Ave		Residential
Cole Ave		Residential
Trewin Ln		Residential
Lawrence Gt / Cres	N-S	Minor Arterial
Hetherington Dr		Residential
Holgate Cres		Residential
Doreen Cres		Residential
Quinn Dr		Residential
The Bridle Path		Residential
Park Ln Circ		Residential
Hillier St		Residential
Rosalynne Ave		Residential
Spry Ave		Residential
Carruthers Dr		Residential
Loscombe Dr		Residential
John Scott Ave		Residential
Lockhart Gt		Residential
Sandringham Dr	W-E	Minor Arterial
Short Cres		Residential
Avondale Dr	N-S	Minor Arterial
Caleche Ave		Residential
Richard Gay Ave		Residential
Stagemaster Cres		Residential
Fieldcrest Ave	N-S	Minor Arterial
Pingle Dr		Residential
Farmington Dr		Residential
Stonefield St		Residential

Name of Road / Highway / Station	Direction	Road Type
Wilkins Cres		Residential
Brownstone Cres		Residential
Hearthstone Cres		Residential
Weaver St		Residential
Phair Ave		Residential
Stirling Ave		Residential
Kennedy Dr		Residential
Faircomb Cres		Residential
McMann Cres		Residential
Strahallan Dr	W-E	Minor Arterial
Bushford St		Residential
Buyson Cres		Residential
Poolton Cres		Residential
Stuart Rd		Residential
Stephen Ave		Residential
Lyndale Cres		Residential
Claret Rd		Residential
Windham Cres		Residential
Parklawn Dr		Residential
Hillhurst Cres		Residential
Inglis Ave		Residential
Yorkville Dr	W-E	Minor Arterial
Granville Dr	N-S	Minor Arterial
Glenabbey Dr	W-E	Minor Arterial
Beechnut Cres		Residential
Rex Tooley Ln		Residential
Oke Rd		Residential
John Walter Cres		Residential
William Ingles Dr		Residential
Wade Sq		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Adair St		Residential
Katerson Ln		Residential
Meadowglade Rd	W-E	Minor Arterial
Worden Dr		Residential
Hayman St		Residential
Cameron Ferguson St		Residential
Arnold Johnston St		Residential
Old Kingston Rd	W-E	Minor Arterial
Osgoode Gt		Residential
Robert Adams Dr	N-S	Minor Arterial
Renwick Rd		Residential
White Cliffe Dr		Residential
Halstead Rd		Residential
Hathaway Dr		Residential
Decoe Crt		Residential
Mulholland Crt		Residential
Worthington Dr		Residential
Sagewood Ave		Residential
Thornbury St		Residential
Saddlebrook Crt		Residential
Glen Eagles Dr		Residential
Pears Crt		Residential
Sheenan Crt		Residential
Hampstead Gt		Residential
Cale Ave		Residential
McRoberts Cres		Residential
Ferris Sq		Residential
Huntington Cres		Residential
Shuttleworth Dr		Residential

Name of Road / Highway / Station	Direction	Road Type
Partner Dr		Residential
Beckett Cres		Residential
Auburn Ln		Residential
Hemmingway Dr		Residential
Bruntsfield St		Residential
Newport Ave		Residential
Pebble Beach Dr		Residential
Pinedale Cres		Residential
Summerlea Crt		Residential
Turnberry Cres		Residential
Darlington Blvd	N-S	Minor Arterial
Foxhunt Tr		Residential
Empire Cres		Residential
Kingsview Crt		Residential
Edinburgh Ln		Residential
Kingswood Dr		Residential
Kingsway Gt		Residential
Barron Crt		Residential
Olive Ave	W-E	Arterial
Birkdale Crt		Residential
Sunnybrae Cres		Residential
Cherrydown Dr	W-E	Minor Arterial
Pinehurst Ave		Residential
Sunningdale Ave		Residential
Capilano Cres		Residential
Annandale St		Residential
Augusta Crt		Residential
Glenridge Crt		Residential
Labrador Dr		Residential
McClure Crt		Residential
Athabasca St	N-S	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Sutton Ave		Residential
Maclaren St		Residential
Erinlea Ave		Residential
Wakefield Cres		Residential
Eastlawn St	N-S	Minor Arterial
Merivale Crt		Residential
Carling Ave		Residential
Winter Ave		Residential
Mackenzie Ave		Residential
Kingsmere Ave		Residential
Belvedere Ave		Residential
Lisgar Ave		Residential
Thornccliffe St		Residential
Ridgecrest Ave		Residential
Gatineau St		Residential
Eton St		Residential
Windermere St		Residential
Cumberland Crt		Residential
Ellesmere Crt		Residential
Springdale Crt		Residential
Keewatin St S	N-S	Minor Arterial
Oriole Crt		Residential
Applegrove Ave		Residential
Oriole St		Residential
Melrose St		Residential
Basswood Ave / Crt		Residential
Viewmount St		Residential
Palm Crt		Residential
Hawthorne Crt		Residential
Lorindale Dr	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
Ivy Crt		Residential
Martindale St		Residential
Oakdale Dr		Residential
Queensdale Ave		Residential
Walnut Crt		Residential
Carnation Crt		Residential
Capri Crt		Residential
Florell Dr	N-S	Minor Arterial
Harcourt Dr		Residential
Dianne Dr	N-S	Minor Arterial
Karen Crt		Residential
Brenda Crt		Residential
Susan Crt		Residential
Denise Dr		Residential
Ronlea Ave		Residential
Carolyn Ave		Residential
Cherryhill St		Residential
St Andrews St		Residential
Augusta Ave		Residential
Palace St	W-E	Minor Arterial
Brunswick St / Crt		Residential
Riverside Dr N/S	N-S	Minor Arterial
Hoskin Ave		Residential
Taylor Ave	W-E	Minor Arterial
Poplar St / Crt		Residential
Linden St / Crt		Residential
Elmridge St		Residential
Wicklow Dr		Residential
Chesterton Ave		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Browning St		Residential
Shelley Ave		Residential
Tennyson Ave / Crt		Residential
Milton St		Residential
Emerson Ave / Crt		Residential
Coleridge St		Residential
Whitman Cres		Residential
Dean Ave	W-E	Minor Arterial
Addison Cres		Residential
Carman Crt		Residential
Shakespeare Ave	W-E	Minor Arterial
Byron Crt		Residential
Keates Ave		Residential
Chaucer Ave		Residential
Macaulay St		Residential
Loring St		Residential
Austen Crt		Residential
Guelph St	N-S	Minor Arterial
Baldwin St	N-S	Minor Arterial
Windsor St	N-S	Minor Arterial
Crerar Ave	W-E	Minor Arterial
Gliddon Ave	W-E	Minor Arterial
Devon Ave		Residential
Athol St E	W-E	Minor Arterial
Highland Ave	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
Cadillac Ave N / S	N-S	Minor Arterial
Lasalle Ave	N-S	Minor Arterial
Central Park Blvd N/S	N-S	Minor Arterial
Arthur St	W-E	Minor Arterial
Bruce St	W-E	Minor Arterial
Oshawa Blvd N/S	N-S	Minor Arterial
Rowe St		Residential
Eulalie Ave	W-E	Minor Arterial
Festhubert Ave		Residential
Courcellette Ave		Residential
Vimy Ave		Residential
Verdun Rd		Residential
St Eloi Ave		Residential
Chadburn Crt		Residential
Mitchell Ave	W-E	Minor Arterial
Viola St		Residential
Kitchener Ave		Residential
Monsah Ave		Residential
Currie Ave		Residential
Montgomery St		Residential
Christine Cres		Residential
Nevis Ave		Residential
Normandy St		Residential
Lomond St		Residential
Dieppe Ave / Crt		Residential
Sterling Ave		Residential
Hillcrest Dr		Residential
Dunkirk Ave		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Sedan Cres		Residential
Brest Crt		Residential
Drew St	N-S	Minor Arterial
Huron St	N-S	Minor Arterial
Charles St	N-S	Minor Arterial
Court St		Residential
Mary St N / S	N-S	Arterial
Albert St	N	Minor Arterial
Celina St	S	Minor Arterial
John St W / E	W-E	Minor Arterial
Emma St	W-E	Minor Arterial
Hogarth St		Residential
Wilkinson Ave		Residential
Elm St	W-E	Minor Arterial
Maple St		Residential
Banting Ave	W-E	Minor Arterial
Barrie Ave	W-E	Minor Arterial
McKim St		Residential
Summer St		Residential
Stacey Ave	W-E	Minor Arterial
Tylor Cres		Residential
George St		Residential
Edward Ave		Residential
Graburn Ave		Residential
Beatty Ave		Residential
McNaughton Ave		Residential

Name of Road / Highway / Station	Direction	Road Type
Etna Ave		Residential
Toronto Ave		Residential
Jackson Ave		Residential
Howard St	N-S	Minor Arterial
First Ave	W-E	Minor Arterial
Lviv Blvd		Residential
Third Ave		Residential
Front St	N-S	Minor Arterial
Elena Ave		Residential
Albany St		Residential
Fisher St		Residential
Ray St	N-S	Minor Arterial
Ontario St	N-S	Minor Arterial
Richmond St E	W-E	Minor Arterial
Colborne St E	E	Minor Arterial
Brock St E	W	Minor Arterial
Elgin St E	W-E	Minor Arterial
Dearborn Ave		Residential
Kendal Ave		Residential
Carriage Works Dr	N-S	Minor Arterial
William St E	W-E	Minor Arterial
Divison St	N-S	Minor Arterial
Agnes St		Residential
kenneth Ave	N-S	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Roxborough Ave	N-S	Minor Arterial
Patricia Ave		Residential
Delroy Crt		Residential
Westminister Ave		Residential
Beverly St		Residential
Luke St		Residential
Oakes Ave		Residential
Lasalle Crt		Residential
Rogers St		Residential
Dover St		Residential
Digby Ave		Residential
Surrey Dr		Residential
Coventry Crt		Residential
Landsdowne Dr		Residential
Sussex St		Residential
Claymore Cres		Residential
Cambridge Ave		Residential
Regent Dr	W-E	Minor Arterial
Eastglen Dr		Residential
Easthaven St		Residential
Florian Crt		Residential
Eastgrove Ave		Residential
Eastdale Ave		Residential
Eastbourne Ave		Residential
Ascot Crt		Residential
Arden Dr / Crt		Residential
Acadia Dr		Residential
Eastmount St		Residential
Parklane Ave		Residential
Woodlane Crt		Residential
Baker Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Beaufort Ave / Crt		Residential
Southwood St		Residential
Conifer St		Residential
Cherry St		Residential
Holly Crt		Residential
Cleta Crt		Residential
Briar Crt		Residential
Laurel Crt		Residential
Heather Crt		Residential
Newbury Ave		Residential
Grandview St N	N-S	Minor Arterial
Cardinal Crt		Residential
Bluefinch Crt		Residential
Blue Heron Dr		Residential
Killdeer Dr		Residential
Bluejay Cres		Residential
Norwood Crt		Residential
Fleetwood Dr		Residential
Eldorado Ave		Residential
Belair Cres		Residential
Kingsway College Dr		Residential
Rockcliffe St		Residential
Maracle Rd		Residential
Violet Hall Rd		Residential
Clarence Biesenthal Dr		Residential
Leland Rd		Residential
Wilbert Bresett Rd		Residential
Wagar Crt		Residential
Shankel Rd		Residential
Bradenton Path		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Apollo St		Residential
Malibu St		Residential
Wood St		Residential
Rolson St		Residential
Haig St		Residential
French St		Residential
Jarvis St	N-S	Minor Arterial
Kingsdale Ave / dr		Residential
Leslie Ave		Residential
Aberdeen St		Residential
Masson St	N-S	Minor Arterial
Leslie St		Residential
Rosedale Ave	W-E	Minor Arterial
Grove Ave		Residential
Sutherland Ave		Residential
Connaught St		Residential
Hillcroft St	W-E	Minor Arterial
Adeline Ave		Residential
Trick Ave		Residential
Pearson St		Residential
Greta St	W-E	Minor Arterial
Grierson St		Residential
Minto St / Crt		Residential
Hillsdale Ave		Residential
Laracor Ln		Residential
Jasmine Cres		Residential
Lilac Crt		Residential
Tulip Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Darcy St	W-E	Minor Arterial
Juniper St / Crt		Residential
Violet Crt		Residential
Verbana Crt		Residential
Wildflower Crt		Residential
Marigold Ave / Crt		Residential
Robert St	W-E	Minor Arterial
Gardenia Crt		Residential
Orchid Crt		Residential
Lavender Crt		Residential
Marica Ave		Residential
Caledon Crt		Residential
Spirea Crt		Residential
Sycamore Cres		Residential
Iris Crt		Residential
Trillium Crt		Residential
Beatrice St E	W-E	Minor Arterial
Lobelia Crt		Residential
Nonquon Rd		Residential
Pentland St		Residential
Lauder Rd		Residential
Maplewood Dr		Residential
Orange Cres		Residential
Juliana Dr		Residential
Bernhard Cres		Residential
Amstel Cres		Residential
Marken Cres		Residential
Arnhem Dr		Residential
Holcan Ave		Residential
Fernwood Ave		Residential
Rembrandt Crt		Residential



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Crestwood Dr		Residential
Everson Crt		Residential
Oakwood Ave		Residential
Brentwood Ave		Residential
Edgewood Ave		Residential
Beechwood St		Residential
Pinewood St		Residential
Dogwood Ave		Residential
Harwood Dr		Residential
Humewood Ave		Residential
Wychwood St		Residential
New Gate Ave		Residential
Clifton Dr		Residential
Rodney Crt		Residential
Lexington St		Residential
Exeter St	N-S	Minor Arterial
Mayfair Ave		Residential
Terrace Dr	N-S	Minor Arterial
Canonberry Crt		Residential
Ashley Crt		Residential
Hackney Crt		Residential
Carnaby Crt		Residential
William Booth Cres		Residential
Lambeth Crt		Residential
Charrington Ave		Residential
Whitehall Crt		Residential
Downing Crt		Residential
Tiffany Circ		Residential
Paddington Cres		Residential
Old Brampton Crt		Residential
Chelsea Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Old Pye Crt		Residential
Torrington Crt		Residential
Trowbridge Crt		Residential
Highgate Ave		Residential
Burnley Crt		Residential
Cardigan Crt		Residential
Compton Cres		Residential
Kensington Cres		Residential
Trowbridge Dr		Residential
Dover St		Residential
Brighton Crt		Residential
Aspen Crt		Residential
Gothic Crt / Dr		Residential
Greenbriar Dr		Residential
Grange Crt		Residential
Camelot Crt / Dr		Residential
Chancery Crt		Residential
Gaylord Dr		Residential
Merlin Crt		Residential
Percival Crt		Residential
Cavendish Crt		Residential
Lancelot Cres		Residential
Gentry Cres		Residential
Glebe Ave		Residential
Galahad Dr		Residential
Gladfern St		Residential
Pascoe Crt		Residential
Avery Crt		Residential
Deauville Crt		Residential
Attersley Dr		Residential
Bayla Crt		Residential
Foxrun Crt		Residential
Cricklewood Dr		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Cobblehill Dr		Residential
Courville Crt		Residential
Bennett Cres		Residential
Mountjoy Crt		Residential
Hayes Ave		Residential
Lavis St / Crt		Residential
Storie Ave		Residential
Dyer Crt		Residential
Crowells St / Crt		Residential
Meadowhill Crt		Residential
Trailridge Cres		Residential
Cresthill Crt		Residential
Strawberry Crt		Residential
Pepperbush Crt		Residential
Elderberry Dr		Residential
Idylwood Crt		Residential
Greenlane Dr / Crt		Residential
Pondtail Crt		Residential
Beaconhill Crt		Residential
Snowberry St / Crt		Residential
Wolfberry Crt		Residential
Buttonbush Crt		Residential
Keswick Crt		Residential
Greystone Crt		Residential
Brasswinds Tr		Residential
Songbird Dr		Residential
Cascade Dr		Residential
Summerwood Hgts		Residential
Silverfox Crt		Residential
Grand Ridge Ave		Residential
Taggart Cres		Residential

Name of Road / Highway / Station	Direction	Road Type
Langley Circ / Gt		Residential
Walter Ave		Residential
Blackthorn St		Residential
Nina Crt		Residential
Cranberry St		Residential
Pinetree Crt		Residential
Thimbleberry Circ		Residential
Palmtree Cres		Residential
Lemans Ave		Residential
Safari Dr		Residential
Century St		Residential
Skylark Ave		Residential
Laguna St		Residential
Corsica Ave		Residential
Astra Ave		Residential
Le Sabre St		Residential
Andover Crt / Dr		Residential
Vega St		Residential
Nova St		Residential
Kilmaurs Ave / Crt		Residential
Dartmoor St		Residential
Hartgrove Ln		Residential
Aldershot Dr		Residential
Faywood Cres		Residential
Margate Dr		Residential
Nottingham Cres		Residential
Langford St		Residential
Shaftsbury St		Residential
Oldman Rd		Residential
Cotsworld Crt		Residential
Dickers Dr		Residential
Traddles Ave		Residential
Wickham St		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Micawber St		Residential
Peggotry Circ		Residential
Copperfield Dr		Residential
Steerforth St		Residential
Coyston Crt / Dr		Residential
Beneford Rd		Residential
Jim Brewster Circ		Residential
Drinkle Cres		Residential
Wadebridge Cres		Residential
Autumnwood Tr		Residential
Kettering Dr		Residential
Krawchuk Cres		Residential
Oxbow Cres		Residential
Aldsworth Cres		Residential
Cronk Crt		Residential
Hanmore St / Crt		Residential
Baynes Ave		Residential
Maddock Dr / Crt		Residential
Corbetts Rd		Residential
Grandlea Crt		Residential
Ripley Cres		Residential
Kingsley Crt		Residential
Lindsay Blvd		Residential
Sproule Cres		Residential
Stone Cottage Cres		Residential
Royal Orchard Dr		Residential
Ridge Valley Dr		Residential
Sandcliff Dr		Residential
Rathburn St		Residential
Trail Valley Dr		Residential
Pondview Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Edward Bolton Cres		Residential
Tall Pine Crt		Residential
Glenbourne Dr / St / Crt		Residential
Glaspell Cres		Residential
Gyatt Cres		Residential
Whitelaw Ave		Residential
Stire St		Residential
Meath dr		Residential
Magnolia Ave		Residential
Ashgrove Cres		Residential
Liveoak St		Residential
Ridgemount Blvd		Residential
Macinally Crt		Residential
Benson St		Residential
Mountview Dr / Crt		Residential
Highbrooke Crt		Residential
Summitview Cres		Residential
Forest Hill Crt		Residential
Springbank Dr		Residential
Westridge Dr / Crt		Residential
Roseheath St		Residential
Hinterland Crt		Residential
Swiss Hgts		Residential
Matterhorn St		Residential
Oberland Dr		Residential
Interlake Dr		Residential
William Tell Dr		Residential
Briarwood Dr		Residential
Pinecrest Rd		Residential
Bridle Crt		Residential
Varcoe Rd		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Dale Park Dr		Residential
Dalepark Dr		Residential
Cherry Blossom Cres		Residential
Briar Hill Gate		Residential
Valleycrest Dr		Residential
Centrefield Dr		Residential
Bellevue Crt		Residential
Windsor Valley Pl / Gt		Residential
Black Creek Tr		Residential
Carriage Ln		Residential
Barrington Pl		Residential
Nash Rd	W-E	Minor Arterial
Lawson Rd		Residential
Wabbokish Crt		Residential
Sheco Crt		Residential
Cloverfield St		Residential
Washburn Park		Residential
Spyfield Tr		Residential
Tooley Rd	N-S	Minor Arterial
Rowland Crt		Residential
McLellan Dr		Residential
Oban Crt		Residential
Alderbrook Dr		Residential
Goldpine Ave		Residential
Abbeywood Cres		Residential
Mossgrave Crt		Residential
Devondale St		Residential
George Reynolds Dr		Residential
Mull Cres		Residential

Name of Road / Highway / Station	Direction	Road Type
Birchfield Dr		Residential
Centrefield Dr		Residential
Homefield Sq		Residential
Oakfield Gt		Residential
Hartsfield Dr		Residential
Old Varcoe Rd		Residential
Mahaffy Pl		Residential
Springfield Ln		Residential
McLean Rd		Residential
Longwood Crt		Residential
Broadlands Cres		Residential
Firwood Ave		Residential
Kintyre St		Residential
Dunkin Ave		Residential
Arran Crt		Residential
Leith Crt		Residential
Jura Crt		Residential
Islay Crt		Residential
Mallory St		Residential
Daiseyfield Ave		Residential
Page Pl		Residential
Adelaide Ave		Residential
Niddery St		Residential
Vetzel Crt		Residential
Vivian Dr		Residential
Timberlane Crt		Residential
Sherry Ln		Residential
Prince Rupert Dr		Residential
Lord Duncan Crt		Residential
Firmer St		Residential
Fices Rd		Residential
Richfield Sq		Residential
Westmore St		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Lynwood Ave		Residential
Glenview Rd		Residential
Fourth Ave		Residential
Jane Ave		Residential
Sleeman Sq		Residential
Cecil Found Cres		Residential
Pidduck St		Residential
Meredith Crt		Residential
Skinner Crt		Residential
Pebblestone Rd	W-E	Minor Arterial
Tyler St		Residential
Leith Crt		Residential
Bradley Blvd		Residential
Progress Dr		Residential
Fewster St		Residential
Jolliffe St		Residential
Living Crt		Residential
Moyse Dr		Residential
Moulton Crt		Residential
Simnick Cres		Residential
Harry Gay Dr		Residential
Duval St		Residential
Tabb St		Residential
Elmer Adams Dr		Residential
Holyrod Dr		Residential
Arthur Trewin St		Residential
Gordon Cowling St		Residential
Brookhill Blvd		Residential
Meachin gt		Residential

Name of Road / Highway / Station	Direction	Road Type
Hovey Ln		Residential
Ted Miller Cres		Residential
Daigle Ln		Residential
Purdy Pl		Residential
Quick Tr		Residential
Murray Tabb St		Residential
Harvey Jones Ave		Residential
Summersford Dr		Residential
Gough Ln		Residential
Carl Raby St		Residential
Forsey Ln		Residential
Ross Wright Ave		Residential
Kilpatrick Crt		Residential
Stevens Rd	W-E	Minor Arterial
Uptown Ave		Residential
Old Scugog Rd	N-S	Minor Arterial
Buttery Crt		Residential
Maryleah Crt		Residential
Taunus Crt		Residential
Glenelge Crt		Residential
Craig Crt		Residential
Munday Crt		Residential
Wellington St		Residential
Sturrock Ave		Residential
Rehder Ave		Residential
Edsall Ave		Residential
Frederick Ave		Residential
Luvmere Crt		Residential
Linden Ln		Residential
Barbara St		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Jackman Rd		Residential
Don Morris Crt		Residential
Mill Ln		Residential
West Scugog Ln		Residential
Terry Cres		Residential
Willoughby Pl		Residential
Kaukonen Crt		Residential
Crockett Pl		Residential
N Scugog Crt		Residential
Westover Dr		Residential
Piper Cres		Residential
Hockley Ave		Residential
Nicks St		Residential
Childs Crt		Residential
Bons Ave		Residential
Lunney Cres		Residential
Goddall Cres		Residential
Dan Sheehan Ln		Residential
Edwin Carr St	N-S	Minor Arterial
Kenneth Cole Dr		Residential
Carey Ln		Residential
Richard Davies Cres		Residential
Robb Ln		Residential
Sidney Rundle Ave		Residential
Northglen Blvd	W-E	Minor Arterial
Loana Ln		Residential
Jerome Way		Residential
Moses Cres		Residential
Crombie St		Residential

Name of Road / Highway / Station	Direction	Road Type
John Matthew Cres		Residential
Jack Roach St		Residential
Ray Richards St		Residential
Fred Jackman Ave		Residential
William Fair Dr		Residential
Bruce Cameron Dr		Residential
Arthur McLaughlin St		Residential
Henry Smith Ave		Residential
Temperance St	N-S	Minor Arterial
Silver St		Residential
Brown St		Residential
Church St	W-E	Minor Arterial
Horsey St		Residential
Beech Ave		Residential
Lowe St		Residential
Liberty Pl		Residential
Carlisle Ave		Residential
Centre St		Residential
Grants Ln		Residential
Alexander Blvd		Residential
Lovers Ln		Residential
Concession St W / E	W-E	Minor Arterial
O'Dell St		Residential
Prospect St	N-S	Minor Arterial
High St		Residential
Burk Crt		Residential
Borland Crt		Residential
Saunders Crt		Residential

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Lorraine Crt		Residential
Prout Dr		Residential
Lambs Ln		Residential
Elgin St E		Residential
First St		Residential
2nd St		Residential
3rd St		Residential
Bernard St		Residential
Summerfield Crt		Residential
Sunset Rd		Residential
Vanstone Crt		Residential
Sunicrest Crt		Residential
Veterans Ave		Residential
4th St		Residential
Hilltop Dr		Residential
Shoreview Dr / Crt		Residential
Meadowview Blvd		Residential
Aldcroft Cres		Residential
Clayton Cres		Residential
Argent St		Residential
Longworth Ave	W-E	Minor Arterial
Daley Ave		Residential
Hogan cres		Residential
Markham Tr		Residential
Streathern Way		Residential
Ken Bromley Lane		Residential
Somerscales Dr		Residential
Laurelwood St		Residential
Willey Dr		Residential
Birmingham Ave		Residential
Goodwin Ave		Residential

Name of Road / Highway / Station	Direction	Road Type
Honeyman Dr		Residential
Darryl Caswell Way		Residential
Allworth Cres		Residential
Allison St		Residential
Lander Cres		Residential
Colville Ave		Residential
Wyse Gt		Residential
Gimblett St		Residential
Courtney St		Residential
Brough Crt		Residential
McCorkell St		Residential
Jennings Dr		Residential
Keeler Cres		Residential
David Baker Crt		Residential
Bavin St		Residential
Higbee Ln		Residential
Ambereen Pl		Residential
Concession Road 3	W-E	Minor Arterial
Northglen Blvd	N-S	Minor Arterial
John Stalker Dr		Residential
Harry Lee Cres		Residential
Higham Pl		Residential
Rebecca Crt		Residential
Pamela Crt		Residential
Avi Crt		Residential
Sydel Crt		Residential
Gary Crt		Residential
Middle Rd	N-S	Minor Arterial
Concession Road 4	W-E	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Liberty St N	N-S	Arterial
Scugog St	N-S	Arterial
Soper Crt		Residential
Hobbs Dr		Residential
Duke St		Residential
Wharf St		Residential
Simpson Ave		Residential
Mearns Crt		Residential
Caristrap St		Residential
Lambs Rd	N-S	Minor Arterial
Haines St	N-S	Minor Arterial
Parkway Ave / Cres		Residential
Flett St		Residential
Southway Dr		Residential
Nelson St		Residential
Morgandale Cres		Residential
Deerpark Cres		Residential
Jane St		Residential
Wilde Crt		Residential
Hailey Crt		Residential
Ashdale Cres		Residential
Prince St		Residential
Frank St		Residential
Queen St	W-E	Minor Arterial
Mearns Ave	N-S	Minor Arterial
Lambert St		Residential
Church St		Residential
Kingscourt Rd		Residential
Galbraith Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Climie Crt		Residential
Royal Pines Crt		Residential
Orchard Park Dr		Residential
Peachtree Cres		Residential
Strathmanor Dr		Residential
Merryfield Crt		Residential
Trudeau Dr		Residential
Marchwood Cres		Residential
Orr Cres		Residential
Hendy Gt		Residential
Dadson Dr		Residential
Squire Fletcher Dr		Residential
McFeeters Cres		Residential
Clinton Crt		Residential
Soper Creek Dr		Residential
Downham Dr		Residential
Souch Crt		Residential
Barley Mill Cres		Residential
Farncomb Cres		Residential
Herriman St		Residential
Mann St		Residential
Tucker Rd		Residential
Baker Crt		Residential
Apple Blossom Blvd		Residential
Glanville Cres		Residential
Tilley Rd		Residential
Bradshaw St		Residential
Maconnachie Pl		Residential
Kershaw St		Residential
Chance Crt		Residential
Edgerton Dr		Residential



NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Swindells St		Residential
Flaxman Ave		Residential
Forrester Dr		Residential
Redfern Cres		Residential
Elephant Hill Dr		Residential
Ireland St		Residential
Lyle Dr		Residential
Brent Cres		Residential
Scottsdale Dr		Residential
Assunta Ln		Residential
Courvier Cres		Residential
Quackenbush St		Residential
William Cowles Dr		Residential
Barlow Crt		Residential
Brooking St		Residential
Stephens Gulch Dr		Residential
Eldad Dr		Residential
Rickaby St		Residential
Dart Crt		Residential
Guildwood Dr		Residential
Lownie Crt		Residential
Budd Ln		Residential
Sprucewood Cres		Residential
Hutton Pl		Residential
Madden Pl		Residential
Cotton St		Residential
Taft Pl		Residential
Crough St		Residential
Hanna Dr		Residential
Laprade Sq		Residential
Lobb Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Fenwick Ave		Residential
Freeland Ave		Residential
Hanning Crt		Residential
Elford Dr		Residential
Pomeroy St		Residential
Bates Crt		Residential
Jollow Dr		Residential
Maxwell Crt		Residential
Hooper Sq		Residential
Champine Sq		Residential
Bethesda Rd	N-S	Minor Arterial
Stephen Mills Rd		Residential
Darlington Clarke Townline Rd	N-S	Minor Arterial
Bennett Rd	N-S	Minor Arterial
Baseline Rd E	W-E	Minor Arterial
Rickard Rd	N-S	Minor Arterial
Providence Rd	N-S	Minor Arterial
Bragg Rd	N-S	Minor Arterial
Taunton Rd	W-E	Arterial
Highway 2	W-E	Arterial
Cobbledick Rd	N-S	Minor Arterial
Lovekin Rd	W-E	Minor Arterial
Browview Rd	W-E	Minor Arterial
Gibson Rd		Residential
Pollard Rd	N-S	Minor Arterial

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Name of Road / Highway / Station</b>	<b>Direction</b>	<b>Road Type</b>
Concession Road 5	W-E	Minor Arterial
Hwy 115	N-S	Arterial
Canadian Pacific Railway North of Hwy 401	W-E	Rail
Canadian Pacific Railway South of Hwy 401	W-E	Rail
Oshawa Executive Airport		Airport
Port of Oshawa East Pier		Pier

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

**APPENDIX C – List of Park Spaces and Water Bodies within the Survey Area**

<b>Park Spaces and Water Bodies</b>	<b>Location</b>
Lakeview Park / Beach	Oshawa
Southmead Park	Oshawa
Lake Ontario	Multiple municipalities
Oshawa Creek	Oshawa
Cordova Park	Oshawa
Chopin Park	Oshawa
Eastview Park	Oshawa
Woodview Park	Oshawa
Connaught Park	Oshawa
Centennial Park	Oshawa
Central Park	Oshawa
Northway Court Park	Oshawa
North Oshawa Park	Oshawa
Hyde Park	Oshawa
Bathe Park	Oshawa
Conant Park	Oshawa
Kingside Park	Oshawa
Knights of Columbus Park	Oshawa
Eastbourne Park	Oshawa
Galahad Park	Oshawa
Attersley Park	Oshawa
Swiss Height Park	Oshawa
Iroquois Shoreline Park	Oshawa
Ridge valley Park	Oshawa
Corbett's Park	Oshawa
Harmony Valley Conservation Area	Oshawa
Easton Park	Oshawa
Baker Park	Oshawa

<b>Park Spaces and Water Bodies</b>	<b>Location</b>
Martindale Park	Oshawa
Harmony Village Park	Oshawa
Florell Park	Oshawa
Grandview North / South Park	Oshawa
Second Marsh Wildlife Area	Oshawa
McLaughlin Bay	Oshawa
McLaughlin Bay Wildlife reserve	Oshawa
Rosswell Park	Courtice
Terry Fox Park	Oshawa
"Oshawa Valleylands Conservation Area"	Oshawa
MacKenzie Park	Oshawa
Margate Park	Oshawa
Kettering Park	Oshawa
Pinecrest Park	Oshawa
Glenbourne Park	Oshawa
South Courtice Dog Park	Courtice
Gatehouse Parkette	Courtice
Glenabbey Park	Courtice
Courtice Duck Pond	Courtice
Tooley's Mill Park	Courtice
Courtice West Park	Courtice
Highland Park	Courtice
Penfound Park	Courtice
Bathgate Park	Courtice
Darlington Provincial Park	Bowmanville
Stuart Park	Courtice
Zion Park	Clarington
Avondale Park	Courtice
Alijco Beach	Courtice

NEDO-33951 REVISION 2  
NON-PROPRIETARY INFORMATION

Park Spaces and Water Bodies	Location
W & D Courtice Memorial Park	Courtice
Moyse Parkette	Courtice
Darlington Hydro Soccer Fields	Clarington
Darlington Waterfront Trail	Clarington
Burk Pioneer Cemetery	Clarington
Harvey Jones Park	Bowmanville
Green Park	Bowmanville
Baxter Park	Bowmanville
Baseline Park	Bowmanville
West Side Park	Bowmanville
Landerville Parkette	Bowmanville
Northglen park	Bowmanville
Douglas Kemp Parkette	Bowmanville
Bons Park	Bowmanville
"Bowmanville Valley Conservation Area"	Bowmanville
Rotary Park	Bowmanville
Bowmanville Creek Barrier Dam	Bowmanville
Waverley Park	Bowmanville
"Bowmanville Westside Conservation Area"	Bowmanville
Bowmanville Harbour	Bowmanville
Port Darlington West / East Beach	Bowmanville
Lions Parkette	Bowmanville
Nelson Parkette	Bowmanville
Argent Park	Bowmanville
Barlow Court Parkette	Bowmanville
Elephant Hill Park	Bowmanville

Park Spaces and Water Bodies	Location
Bowmanville Cemetery	Bowmanville
"Bowmanville Soper Creek Playground"	Bowmanville
Guildwood Park	Bowmanville
Stephen Gulch's Conservation Area	Bowmanville
Samuel Wilmont Natural Area	Newcastle
Mearns Park	Bowmanville
Soper Creek Trail	Bowmanville



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33952

Revision 1

March 7, 2023

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 3  
Safety Objectives and Design Rules for  
Structures, Systems and Components**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	3.0	Updated to 300 MWe.
	3.1.6.2	Included full acronym name for Reactor Pressure Vessel for first time use.
	3.1.7.4	Updated acronym use of DL1.
	3.1.7.9.2	Updated use of RPV acronym.
	3.2.1.1	Edited Safety Category 3 wording.
	3.2.1.3	Edited Primary Function wording.
	3.2.1.4	Added details to Delayed Functions.
	3.2.1.6	Updated, added text and added reference to Table 3.2-2.
	3.2.3	Included full acronym name for Design Basis Earthquake and a pointer to Section 3.3-1.
	3.2.3.1	Added CSA N289.3 reference and updated text.
	3.2.4	Updated reference to Table 3.2-3 (from Table 3.2-2).
	3.2	Added new Table 3.2-2 for Safety Class for SSC.
	Acronym List	DGRS and NBC acronyms added.
	3.3	Updated pointer to Subsection 3.3.7.4
	3.3.1 – 3.3.7	Cross-references to Chapter 2 updated as required.
	3.3.1.1, 3.3.1.1.1-3.3.1.1.4	Updated to incorporate bounding information previously documented in Chapter 2, Section 2.7.
	3.3.1.1.6	Updated content on development of dynamic subgrade profiles and included pointer to Subsection 3.5.2.2.
	3.3.2.1 – 3.3.2.5	Updates made to decouple from Chapter 2 and present bounding design parameters.
	3.3.6.1	Removed reference to Chapter 19 for Fire Protection Program.
	3.3.8	References 3.3-12 to 3.3-20 and 3.3-26 and 3.3-28 added.
	Table 3.3-1	Added reference to CSA N289 series for basis.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
	Table 3.3-2 to 3.3-5	Tables added to supplement added bounding information content in 3.3.1.1.1 – 3.3.1.1.4.
	Figure 3.3-1, 3.3-2, 3.3-5, 3.3-12	Figures added to supplement added bounding information content in 3.3.1.1.1 – 3.3.1.1.4.
	3.4.1.1	Removed reference to Chapter 19 for Fire Protection Program.
	3.4.4	Rephrased reference to areas where postulated pipe breaks are excluded to indicate future analyses are required.
	3.4.4.2.2	Edited Location of Postulated Pipe Break subsection.
	3.4.4.2.3	Edited Location of Postulated Pipe Crack subsection.
	3.5.1, 3.5.2.7 and 3.5.4.1	Reference to NEDC-33926P added.
	3.5.2.2	Revised to incorporate bounding information previously documented in Chapter 2, Section 2.7. Additional text added regarding upper bound nominal water table levels.
	3.5.2.2.1	Bounding Equivalent Subgrade Static Profile Subsection updated.
	3.5.2.2.3	Edited to remove content covered in 3.5.2.2.1.
	3.5.4 and 3.5.4.4.1	Updated containment internal structure descriptions included.
	3.5.5.2.1	Pointer to Design Basis Threat subsection revised.
	3.5.5.4.1	Seismic and Extreme Wind sub-heading revised.
	3.5.7	References 3.5-11, 3.5-12 and 3.5-14 through 19 added to supplement Subsection 3.5.2.2 and 3.5.2.2.1 added information.
	Table 3.5-1 and 3.5-2	Tables added to supplement content update in 3.5.2.2.1.
		Minor editorial updates throughout.
	3.6.3.12	Safety Class 1 updated to Safety Category 1
	3.6.7.2.5	Editing Safety Category wording.
	3.9.2	Updated scope for DEC assessments.
	3.9.3.1	Added RD-2.5.2 reference and updated seismic categorization text.



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
	3.9.3.2	Seismic interaction equipment details removed.
	3.9.3.2.1	Seismic test details added and edits made.
	3.9.3.3	Section 3.3.1.3 pointer added.
	3.9.3.5	Renamed from Seismic Margin to Beyond Design Basis Earthquake and updated content.
	3.9.4.1	Revised content and included additional references.
	3.9.4.4.1	Revised content including DBA groupings.
	Minor editorial updates made including Safety Class 1 updated to SC1 throughout	
	Table 3.12-1	SSC classification table updated to include the latest information (Radiation Monitoring Systems, Wide range pool level instrumentation, Leak detection equipment updated).
	Sections 3.13 – 3.18	Appendices 3B – 3G identifying and describing computer software have been updated to align with the latest information. Where there is a discrepancy identified between software version numbers in these appendices and other PSAR chapters, this appendix should be taken as correct.
	Appendix 3C	Title updated.
	Section 3.14	Introduction description of scope edited.
	Appendix 3D	Title updated.
	Section 3.15	Introduction description of scope edited.
	Appendix 3E	Title updated.
	Section 3.16	Introduction description of scope edited.
	Sections 3.16.18 and 3.17.10	Computer code descriptions updated.
	Appendix 3F	Title updated.
	Section 3.17	Introduction description of scope edited.
	Editorial changes made throughout.	

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**ACRONYM LIST**

<b>Acronym</b>	<b>Explanation</b>
AC	Alternating Current
AEF	Annual Exceedance Frequency
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
API	American Petroleum Institute
ARS	Acceleration Response Spectra
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATH	Acceleration Time History
ATWS	Anticipated Transient Without Scram
AWWA	American Water Works Association
BDBA	Beyond-Design Basis Accident
BDBE	Beyond-Design Basis Earthquake
BDBT	Beyond-Design Basis Threat
BE	Best Estimate
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
BWRX	Boiling Water Reactor, 10th Design
CB	Control Building
CAD	Computer-Aided Design
CCF	Common Cause Failure
CAFTA	Computer Aided Fault Tree Analysis
CANDU	CANada Deuterium Uranium
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
CGD	Canadian Geodetic Datum
CIV	Containment Isolation Valve

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
CLE	Checking Level Earthquake
CNSC	Canadian Nuclear Safety Commission
CPR	Critical Power Ratio
CR	Control Room
CRD	Control Rod Drive
CSA	Canadian Standards Association
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBT	Design Basis Threat
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DGRS	Design Ground Response Spectrum
DL3	Defense Line 3
DL	Defense Line
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FE	Finite Element
FIA	Foundation Interface Analysis
FIRS	Foundation Input Response Spectra
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analyses
FPC	Fuel Pool Cooling and Cleanup System
FSF	Fundamental Safety Function
FW	Feedwater
GEH	GE-Hitachi Nuclear Energy
GMRS	Ground Motion Response Spectra

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
GUI	Graphical User Interface
HCLPF	High Confidence of Low Probability of Failure
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated Leak Rate Test
ISRS	In-Structure Response Spectra
LB	Lower Bound
LL	Live Load
LMP	Licensing Modernization Program
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LOPP	Loss of Preferred Power
LR	Lower Realization
LS	Level Switch
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MAPE	Mean Annual Probability of Exceedance
MCA	Main Condenser and Auxiliaries
MCNP	Monte Carlo N-Particle
MCR	Main Control Room
MS	Main Steam

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
NBC	National Building Code of Canada
NPP	Nuclear Power Plant
NSCA	Nuclear Safety and Control Act
NS-DBE	Non-Seismic Design Basis Earthquake
OBE	Operating Basis Earthquake
OLC	Operational Limits and Conditions
OPG	Ontario Power Generation
P&ID	Piping and Instrumentation Diagram
PAM	Post-Accident Monitoring
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PLSA	Plant Services Area
PMF	Probable Maximum Flood
PRA	Probability Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Assessment
QA	Quality Assurance
RAM	Reliability, Availability, and Maintainability
RB	Reactor Building
RBV	Reactor Building Vibration
RCS	Reactor Coolant System
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Seismic Design Category
SDE	Site Design Earthquake
SEI	Structural Engineering Institute
SIL	Safety Integrity level
SIR	Seismic Interface Restraint
SIT	Structural Integrity Test
SMAMP	Structures Monitoring and Aging Management Program
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SRA	Site Response Analysis
SRSS	Square-Root-of-the Sum of the Squares
SSC	Structures, Systems, and Components
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UHRS	Uniform Hazard Response Spectrum
UL	Underwriters Laboratory
UR	Upper Realization
USNRC	U.S. Nuclear Regulatory Commission
V/H	Vertical to Horizontal
ZPA	Zero Period Acceleration

## TABLE OF CONTENTS

<b>3.0</b>	<b>SAFETY OBJECTIVE AND DESIGN RULES FOR STRUCTURES, SYSTEMS, AND COMPONENTS.....</b>	<b>3-1</b>
3.1	General Safety Design Basis .....	3-1
3.1.1	Safety Objectives .....	3-1
3.1.2	Radiation Protection and Radiological Acceptance Criteria.....	3-2
3.1.3	Plant States Considered in the Design Basis .....	3-3
3.1.4	Prevention and Mitigation of Accidents .....	3-5
3.1.5	Fundamental Safety Functions .....	3-5
3.1.6	Defence-in-Depth .....	3-6
3.1.7	Application of General Design Requirements and Technical Acceptance Criteria .....	3-12
3.1.8	Practical Elimination.....	3-17
3.1.9	Safety Margins and Avoidance of Cliff-Edge Effects.....	3-18
3.1.10	Design Approaches for the Reactor Core and for Fuel Storage .....	3-18
3.1.11	Considerations of Interactions Between Multiple Units .....	3-20
3.1.12	Design Considerations for Ageing Management.....	3-20
3.1.13	References.....	3-21
3.2	Classification of Structures, Systems and Components .....	3-26
3.2.1	Safety Classification Background.....	3-26
3.2.2	Safety Classification Process.....	3-29
3.2.3	Seismic Categories .....	3-30
3.2.4	Quality Group .....	3-31
3.2.5	References.....	3-32
3.3	Protection Against External Hazards .....	3-38
3.3.1	Seismic Design .....	3-38
3.3.2	Extreme Weather Conditions .....	3-75
3.3.3	Extreme Hydrological Conditions .....	3-77
3.3.4	Aircraft Crash .....	3-78
3.3.5	Missiles .....	3-78
3.3.6	External Fires, Explosions and Toxic Gases.....	3-80
3.3.7	Other External Hazards .....	3-81
3.3.8	References.....	3-82
3.4	Protection Against Internal Hazards.....	3-120
3.4.1	Internal Fires, Explosions and Toxic Gases.....	3-120

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

3.4.2	Internal Flooding .....	3-124
3.4.3	Internal Missiles .....	3-124
3.4.4	Pipe Breaks.....	3-125
3.4.5	Other Internal Hazards.....	3-135
3.4.6	References.....	3-136
3.5	General Design Aspect for Civil Engineering Works of Seismic Category Buildings and Civil Engineering Structures .....	3-138
3.5.1	General Design Principles for Seismic Category A Structures .....	3-139
3.5.2	Foundations .....	3-144
3.5.3	Containment.....	3-148
3.5.4	Containment Internal Structures .....	3-154
3.5.5	Reactor Building.....	3-156
3.5.6	Robustness Design of Seismic Category A Structures .....	3-160
3.5.7	References.....	3-164
3.6	General Design Aspects for Mechanical Systems and Components.....	3-180
3.6.1	Special Topics for Mechanical Components .....	3-184
3.6.2	Dynamic Testing and Analysis of Systems, Components, and Equipment.....	3-187
3.6.3	Codes for ASME BPVC Section III Division 1, Class 1, 2 and 3 Components, Component Supports and Core Support Structure.....	3-190
3.6.4	Control Rod Drive System .....	3-201
3.6.5	Reactor Pressure Vessel Internals .....	3-202
3.6.6	Functional Design, Qualification and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints .....	3-202
3.6.7	Piping Design .....	3-202
3.6.8	Threaded Fasteners – Codes for ASME BPVC Section III Division 1 Class 1, 2, and 3 .....	3-211
3.6.9	References.....	3-212
3.7	General Design Aspects for Instrumentation and Control Systems and Components.....	3-227
3.7.1	Performance .....	3-227
3.7.2	Design for Reliability .....	3-227
3.7.3	Independence .....	3-227
3.7.4	Qualification .....	3-227
3.7.5	Verification and Validation .....	3-227
3.7.6	Failure Modes .....	3-227



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

3.7.7	Control of Access to Equipment.....	3-228
3.7.8	Quality.....	3-228
3.7.9	Testing and Testability .....	3-228
3.7.10	Maintainability .....	3-228
3.7.11	Identification of Items Important to Safety.....	3-228
3.8	General Design Aspects for Electrical Systems and Components .....	3-229
3.8.1	Redundancy.....	3-230
3.8.2	Independence .....	3-230
3.8.3	Diversity .....	3-230
3.8.4	Controls and Monitoring.....	3-230
3.8.5	Identification .....	3-231
3.8.6	Capacity and Capability of Systems for Different Plant States .....	3-231
3.8.7	External Grid and Related Issues .....	3-231
3.9	Equipment Qualification .....	3-233
3.9.1	Purpose.....	3-233
3.9.2	Scope.....	3-233
3.9.3	Seismic .....	3-233
3.9.4	Environmental Qualification .....	3-240
3.9.5	Electromagnetic Compatibility.....	3-246
3.9.6	Specific Equipment Requirements.....	3-247
3.9.7	References.....	3-249
3.10	In-Service Monitoring, Tests, Maintenance, and Inspections.....	3-253
3.10.1	Safety Design Bases and Requirements .....	3-253
3.10.2	In-Service Monitoring .....	3-253
3.10.3	In-Service Testing .....	3-253
3.10.4	In-Service Maintenance .....	3-254
3.10.5	In-Service Inspection .....	3-254
3.10.6	References.....	3-255
3.11	Compliance with National and International Standards .....	3-256
3.11.1	References.....	3-256
<b>APPENDIX 3A – PRELIMINARY CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS.....</b>		<b>3-257</b>
3.12	Introduction .....	3-257
3.12.1	References.....	3-257

**APPENDIX 3B – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES .....3-274**

3.13	Introduction .....	3-274
3.13.1	ACS SASSI v4 .....	3-274
3.13.2	ANSYS v17 .....	3-274
3.13.3	Ansys LS-Dyna v2021 .....	3-275
3.13.4	SSI-StressCoord v1 .....	3-275
3.13.5	GT STRUDL.....	3-275

**APPENDIX 3C – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF MECHANICAL STRUCTURES, SYSTEMS AND COMPONENTS .....3-276**

3.14	Introduction .....	3-276
3.14.1	ANSYS v17 .....	3-276
3.14.2	PBLE v1 .....	3-276
3.14.3	SIMCENTER 3D Acoustics v2022 .....	3-276
3.14.4	GT STRUDL.....	3-277
3.14.5	HyperMesh.....	3-277
3.14.6	ERSIN v3 .....	3-277
3.14.7	RINEX Computer Program .....	3-277
3.14.8	PDA (Civil) .....	3-278
3.14.9	PIPESTRESS .....	3-278
3.14.10	FLOMASTER v2021.1 .....	3-278
3.14.11	Ansys LS-Dyna v2021 .....	3-278
3.14.12	3KeyMaster v2021 (ICE/Plant Integration Engineering/Systems Engineering).....	3-279

**APPENDIX 3D – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF ELECTRICAL STRUCTURES, SYSTEMS AND COMPONENTS.....3-280**

3.15	Introduction .....	3-280
3.15.1	ETAP v2021.1 (ICE Systems/I&C Tech).....	3-280
3.15.2	LDRA (I&C Tech/ICE Systems) .....	3-280
3.15.3	Quartus II (I&C Tech/ICE Systems) .....	3-280

**APPENDIX 3E – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSES OF STRUCTURES, SYSTEMS AND COMPONENTS – NUCLEAR FUELS .....3-282**

3.16	Introduction .....	3-282
3.16.1	EPRI: Acube v11.....	3-282
3.16.2	EPRI: CAFTA v11 .....	3-282
3.16.3	EPRI: MAAP v5.....	3-283

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

3.16.4	EPRI: PRAQuant v11 .....	3-283
3.16.5	FURST (Core & Fuel) .....	3-283
3.16.6	GTRAC v1 .....	3-283
3.16.7	MACCS v4 .....	3-284
3.16.8	MCNPX v6 .....	3-284
3.16.9	ORIGEN v1 .....	3-284
3.16.10	PANAC v11 .....	3-285
3.16.11	PRIME v3 .....	3-285
3.16.12	RAMP: GALE v3.2 .....	3-285
3.16.13	RAMP: HABIT v2.2 .....	3-286
3.16.14	RAMP: DandD v2.1 .....	3-286
3.16.15	RAMP: GENII v2.10 .....	3-286
3.16.16	RAMP: MILDOS v4 .....	3-286
3.16.17	RAMP: NRC-RADTRAN v6.02.1 .....	3-287
3.16.18	RAMP: PIMAL v4.1.0 .....	3-287
3.16.19	RAMP: TurboFRMAC v2021 11.0.2 .....	3-287
3.16.20	RAMP: VARSKIN v1.0 .....	3-287
3.16.21	SAP4G07P v7 .....	3-288
3.16.22	SCALE v6 .....	3-288
3.16.23	TGBLA v6 .....	3-288
3.16.24	TRACG v4 .....	3-289
3.16.25	SEISM v5 .....	3-289
3.16.26	DECAY v1 .....	3-289
3.16.27	GTRAC v1 .....	3-290

**APPENDIX 3F – COMPUTER PROGRAMS USED IN ENVIRONMENTAL AND  
RADIOLOGICAL ANALYSES SUPPORTING THE DESIGN OF STRUCTURES,  
SYSTEMS AND COMPONENTS.....3-291**

3.17	Introduction .....	3-291
3.17.1	ADDAM Version 1.4.2 .....	3-291
3.17.2	DECAY v1 .....	3-291
3.17.3	RADTRAD (Analytical Methods/ Radiological Analysis) .....	3-292
3.17.4	RAMP: GALE v3.2 .....	3-292
3.17.5	RAMP: HABIT v2.2 .....	3-292
3.17.6	RAMP: DandD v2.1 .....	3-292
3.17.7	RAMP: GENII v2.10 (Analytical Methods/Radiological Analysis).....	3-293

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

3.17.8	RAMP: MILDOS v4 .....	3-293
3.17.9	RAMP: NRC-RADTRAN v6.02.1.....	3-293
3.17.10	RAMP: PIMAL v4.1.0 .....	3-293
3.17.11	RAMP: TurboFRMAC v2021 11.0.2.....	3-294
3.17.12	RAMP: VARSKIN v1.0 .....	3-295
3.17.13	SAP4G07P v7 .....	3-294
3.17.14	SCALE v6 .....	3-294
3.17.15	TGBLA v6 .....	3-295
3.17.16	TRACG v4.....	3-295
3.17.17	IMPACT .....	3-296

**APPENDIX 3G – COMPUTER PROGRAMS USED IN THE DESIGN OF COMPONENTS,  
SYSTEMS AND STRUCTURES IN SAFETY ANALYSES (PRA AND DETERMINISTIC)**  
..... **3-297**

3.18	Introduction .....	3-297
3.18.1	EPRI: Acube v11.....	3-297
3.18.2	EPRI: CAFTA v11 .....	3-297
3.18.3	EPRI: FRANX v11.....	3-298
3.18.4	EPRI: FTRex v1.8 .....	3-298
3.18.5	EPRI: HRA Calculator .....	3-298
3.18.6	EPRI: MAAP v5.....	3-298
3.18.7	EPRI: PRAQuant v11 .....	3-299
3.18.8	ActivePoint HMI/CIMPLICITY 11 .....	3-299
3.18.9	Control ST – ToolboxST Tool .....	3-299
3.18.10	EPRI: SysImp v11 .....	3-299
3.18.11	EPRI: UNCERT v11 .....	3-300
3.18.12	GOTHIC v8 .....	3-300
3.18.13	MACCS v4 .....	3-300
3.18.14	RAMP: NRC-RADTRAN v6.02.1.....	3-300
3.18.15	SCALE v6 .....	3-301
3.18.16	TRACG v4.....	3-301
3.18.17	VTR.LMP .....	3-301

## LIST OF TABLES

Table 3.1-1: Identification of Defence Levels .....	3-23
Table 3.2-1: Safety Category for Functions Based on Defense Line Assignment .....	3-34
Table 3.2-2: Safety Class for SSC .....	3-35
Table 3.2-3: Codes and Standards for Pressure-Retaining Equipment.....	3-36
Table 3.3-1: Seismic Categories and Design Basis of BWRX-300 Structures.....	3-85
Table 3.3-2: Base Case Rock Dynamic Properties.....	3-86
Table 3.3-3: Base Case Engineered Fill and In-situ Soil Dynamic .....	3-87
Table 3.3-4: Rock Layers Kappa Values .....	3-88
Table 3.3-5: Selected Time History Records .....	3-89
Table 3.3-6: Seismic Damping Values for BWRX-300 Structures .....	3-90
Table 3.3-7: Seismic Damping Values for Primary Subsystems.....	3-91
Table 3.3-8: Preliminary Dynamic Loading $\alpha$ , $\beta$ – Damping.....	3-92
Table 3.3-9: Seismic Damping Values for Piping and Equipment .....	3-93
Table 3.3-10: Piping and Equipment Damping Values for All Other Non-Seismic Loadings ..	3-94
Table 3.5-1: As-Built Static Properties for Soil Layers .....	3-168
Table 3.5-2: Summary of Static Rock Properties.....	3-169
Table 3.5-3: Stability Requirements for RB and Containment Common Mat Foundation.....	3-170
Table 3.5-4: Acceptance Criteria for SCCV .....	3-171
Table 3.5-5: Acceptance Criteria for Containment Closure Head.....	3-172
Table 3.5-6: Acceptance Criteria for Other MC Components .....	3-173
Table 3.6-1: Applicable Pressure Boundary Codes and Standards.....	3-214
Table 3.6-2: Load Combinations and Acceptance Criteria .....	3-215
Table 3.6-3: Comparison of Event Frequency to Plant Conditions and Service Loadings....	3-217
Table 3.6-4: Normal Operating Events (DC-1).....	3-218
Table 3.6-5: Test Events (DC-1) .....	3-219
Table 3.6-6: Anticipated Operational Occurrences (DC-2) .....	3-220
Table 3.6-7: Design Basis Accidents (DC-3) .....	3-221
Table 3.6-8: Design Extension Condition (DC-4).....	3-222
Table 3.6-9: Summary of Cycles of Events.....	3-223
Table 3.6-10: Load Combinations and Acceptance Criteria for ASME BPVC Section III Division 1 Class 1 Piping Systems .....	3-224
Table 3.6-11: Load Combinations and Acceptance Criteria for ASME BPVC III Division 1 Class 2 and 3 Piping Systems.....	3-225

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Table 3.6-12: Load Combinations and Acceptance Criteria for Non-Safety Class Power Piping Systems.....	3-226
Table 3.9-1: Harsh Environment Parameter Conditions .....	3-252
Table 3.12-1: Preliminary BWRX-300 Classification List .....	3-259

## LIST OF FIGURES

Figure 3.1-1: Defence-in-Depth – Plant States and Defense lines .....	3-24
Figure 3.1-2: BWRX-300 Safety Strategy Implementation Process .....	3-25
Figure 3.3-1: Shear Wave Velocities for the Bounding In-situ Profile .....	3-95
Figure 3.3-2: Cases Considered for Explicit Considerations of Epistemic Uncertainties .....	3-96
Figure 3.3-3: Shear Wave Velocity and Layer Thicknesses Randomization – BE-BE Case ...	3-97
Figure 3.3-4: Soil Degradation Curves Randomization.....	3-98
Figure 3.3-5: Uniform Hazard Response Spectra at Bedrock Elevation .....	3-99
Figure 3.3-6: Ground Surface Uniform Hazard Response Spectra .....	3-100
Figure 3.3-7: Rock Top Surface Uniform Hazard Response Spectra .....	3-101
Figure 3.3-8: Ground Surface Composite and Epistemic Log-Normal Standard Deviations .....	3-102
Figure 3.3-9: Rock Top Surface Composite and Epistemic Log-Normal Standard Deviations .....	3-103
Figure 3.3-10: Bounding Horizontal Rock Ground Motion Response Spectra .....	3-104
Figure 3.3-11: Bounding Horizontal Performance Based Surface Response Spectra.....	3-105
Figure 3.3-12: Vertical to Horizontal (V/H) Spectral Ratios.....	3-106
Figure 3.3-13: Comparison of Bounding to Updated Ground Motion Design Response Spectra .....	3-107
Figure 3.3-14: Augmented and Smoothed Horizontal and Vertical Rock Design Ground Response Spectra .....	3-108
Figure 3.3-15: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 North .....	3-109
Figure 3.3-16: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 West .....	3-110
Figure 3.3-17: Vertical Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 Vertical .....	3-111
Figure 3.3-18: Normalized Arias Intensity and Power Spectral Density Function for Response Spectrum Matched HWA026 Acceleration Time Histories .....	3-112
Figure 3.3-19: 5% Damped Response Spectra for Response Spectrum Matched HWA026 Acceleration Time Histories .....	3-113
Figure 3.3-20: Logarithmic Mean of Strain-Compatible Shear Wave Velocities .....	3-114
Figure 3.3-21: Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocities .....	3-115
Figure 3.3-22: Logarithmic Mean of Strain-Compatible Damping Ratios.....	3-116
Figure 3.3-23: Logarithmic Standard Deviation of Strain-Compatible Damping Ratios .....	3-117
Figure 3.3-24: Strain-Compatible Shear Wave Velocity and Damping Using 100 Hz Interpolation.....	3-118

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Figure 3.3-25: Subgrade Profiles for Bounding BWRX-300 Seismic Analyses .....	3-119
Figure 3.5-1: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building .....	3-174
Figure 3.5-2: Containment Closure Head Structure Boundary .....	3-175
Figure 3.5-3: Access Hatch Code Jurisdictional Boundary .....	3-176
Figure 3.5-4: Penetrations Jurisdictional Boundary .....	3-177
Figure 3.5-5: Design Procedures for the Containment Closure Head.....	3-178
Figure 3.5-6: Design Procedures for the MC Components .....	3-179



### 3.0 SAFETY OBJECTIVE AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS

This chapter introduces the safety objectives and the Safety Strategy to meet those objectives for the design and construction of the Boiling Water Reactor, 10th Design – 300 MWe (BWRX-300) Small Modular Reactor (SMR) facility at the Darlington site in Ontario, Canada.

Additionally, this chapter describes the methodology for classification of Structures, Systems, and Components (SSC), the design measures for protection against external and internal hazards, the general design aspects, and codes and standards applied to the BWRX-300 design to meet the requirements of the Nuclear Safety and Control Act (NSCA) and associated Canadian Nuclear Safety Commission (CNSC) Regulations and relevant Regulatory Documents.

#### 3.1 General Safety Design Basis

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy. The objective of the Safety Strategy is to establish a design with a high level of safety. This is accomplished through incorporation of design requirements as set forth in CNSC REGDOC-2.5.2, (Reference 3.1-1) which to a large degree are based on the principles set forth in the International Atomic Energy Agency (IAEA) document SSR-2/1 (Reference 3.1-2).

The establishment of the BWRX-300 design basis is achieved through an iterative safety framework wherein the design is implemented to meet defined safety objectives and safety goals that are confirmed via deterministic and probabilistic safety analyses. Results of safety analyses then provide feedback into the design and the process is repeated as required until adequate design and regulatory safety margins are achieved.

##### 3.1.1 Safety Objectives

In CNSC REGDOC-2.5.2 Section 4 (Reference 3.1-1), the CNSC endorses the safety objectives established by the IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles (Reference 3.1-3) which when followed ensure that reactor facilities are operated, and activities conducted to achieve the highest standards of safety that can be reasonably achieved. These safety objectives are described below:

**General Nuclear Safety Objective:** Reactor facilities are designed and operated in a manner that will protect individuals, society, and the environment from harm by establishing and maintaining effective defences against radiological hazards due to ionizing radiation. The general nuclear safety objective is supported by the following three complementary safety objectives:

1. **Radiation Protection Objective:** Radiation exposures within the reactor facility during normal operations, during anticipated operational occurrences or due to any planned release of radioactive material from the reactor facility are kept below prescribed limits and As Low As Reasonably Achievable (ALARA). Provisions are made for the mitigation of the radiological consequences of accidents.
2. **Technical Safety Objective:** All reasonably practicable measures are taken to prevent accidents in the reactor facility and to mitigate the consequences of events should they occur.
3. **Environmental Protection Objective:** All reasonably practicable mitigation measures to protect the environment during the operation of a reactor facility and to mitigate the consequences of an accident are provided. The design includes provisions to control, treat and monitor releases to the environment and minimize the generation of radioactive and hazardous wastes.

The high-level safety objectives inform the principal safety objectives in the design and safety analyses.

### **3.1.2 Radiation Protection and Radiological Acceptance Criteria**

#### **3.1.2.1 Radiation Protection**

The BWRX-300 is designed to meet the Radiation Protection Objective by ensuring that potential radiation dose to workers and the public is kept below prescribed regulatory limits per the Radiation Protection Regulations (Reference 3.1-4) and ALARA.

This is achieved by a comprehensive and appropriately conservative source term derivation identifying radiation sources during the design phase to ensure means are provided to reduce occupational exposure during plant operation, maintenance, and decommissioning.

Safety features and measures include:

- Passive engineered safety features
- Active engineered safety features
- Administrative safety measures

Engineered safety features include shielding, containment, ventilation, remote handling, and interlocks. Administrative safety measures that reduce exposure to the hazard during planned operations include restrictions on occupancy, monitoring arrangements, pre-planning of exposure and the use of barriers and notices. Passive engineered safety measures (e.g., containment or shielding) are preferred before active engineered safety features and administrative safety measures. Human factors considerations are incorporated into the engineered and administrative measures (See Chapter 18 for details).

System design evaluations are performed in parallel with other activities to ensure systems support operational objectives. These evaluations include the development of reasonable and practical measures to achieve minimal dose to workers and the public.

Details on how radiation protection is considered in the design for operational states and accident conditions are provided in Chapter 12.

#### **3.1.2.2 Radiological Acceptance Criteria**

Limits on radiation dose are established by the CNSC through the Radiation Protection Regulations (Reference 3.1-4). The expectation established is that during normal operation, including maintenance and decommissioning, dose to workers and the public are ALARA.

Per CNSC Radiation Protection Regulations (Reference 3.1-4), the effective dose limit for a nuclear energy worker is an average of 20 mSv effective dose per year over a five-year period (100 mSv over five consecutive years), with no single year exceeding 50 mSv effective dose. The effective dose limit for a member of the public is 1 mSv per year from all sources of radiation other than natural background and medical exposures. Additional details are provided in Reference 3.1-4.

In addition to design features, administrative measures such as radiation protection and environmental protection programs are established to ensure worker and public dose is maintained below limits. Action levels are established for effluent releases and expressed in a form that compliance can be demonstrated in a practical manner. These action levels are not limiting but, are values at which actions must occur to reduce the effluent releases from the plant. Chapter 20 discusses Effluent Dose Levels to the General Public.

Deterministic safety analyses are conducted in accordance with CNSC REGDOC-2.4.1 (Reference 3.1-5) to confirm that the BWRX-300 is designed to ensure that potential radiation doses to the public from Abnormal Operating Occurrences (AOOs) and Design Basis Accidents (DBAs) (defined in Subsection 3.1.3) do not exceed dose acceptance criteria per Section 4.2.1 of CNSC REGDOC-2.5.2 (Reference 3.1-1). In the deterministic safety analysis, the committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated for a period of 30 days after the analyzed event to confirm that for AOOs and DBAs, doses are less than or equal to the following:

- 0.5 millisievert (mSv) for any AOO or
- 20 mSv for any DBA

Chapter 15, Subsection 15.3.1, describes the dose calculation methodology used in the deterministic safety analysis. Results of the analyses are summarized in Section 15.7 demonstrating that the radiological consequences of the analyzed events do not exceed the acceptance criteria for AOOs and for DBAs.

### 3.1.2.3 Safety Goals

In addition to the deterministic dose acceptance criteria, Probabilistic Safety Analysis (PSA) is used to assess risks posed by reactor facility operation through the application of quantitative safety goals. These include core damage frequency, and small and large release frequency.

Core damage frequency is a measure of the capability of the design to prevent an accident that leads to core damage. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of reactor facilities. The quantitative goals as established by CNSC REGDOC-2.5.2, Section 4.2.2 (Reference 3.1-1) are:

- **Core damage frequency** - The sum of frequencies of all fault sequences that can lead to significant core degradation shall be less than 1E-5 per reactor-year.
- **Small release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E15 becquerels of Iodine-131, shall be less than 1E-5 per reactor-year.
- **Large release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E14 becquerels of Cesium-137 shall be less than 1E-6 per reactor-year.

The PSA is described in detail in Chapter 15, Section 15.6, Probabilistic Safety Analyses.

### 3.1.3 Plant States Considered in the Design Basis

The range of conditions and events considered are categorized into plant states based on their frequency of occurrence. Plant states include operational states and accident conditions. Operational states included in the design basis are Normal Operation and AOOs. Accident conditions considered in the design basis are DBAs. Design Extension Conditions (DECs) are accident conditions considered in the design but are outside of the design basis based on their lower expected frequency of occurrence.

These four plant states considered in the BWRX-300 Safety Strategy as described below are consistent with CNSC REGDOC-2.5.2, Section 7.3 (Reference 3.1-1):

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- **Normal Operation** is operation within specified Operational Limits and Conditions (OLCs) (see Chapter 16) and includes the following Normal Plant Operational Modes: Power Operation, Startup, Hot Shutdown, Stable Shutdown, Cold Shutdown, and Refueling. (The normal plant operating modes are described in Chapter 16).
- **Anticipated Operational Occurrences** are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility but that, with the appropriate design measures, do not cause any significant damage to safety class components, or lead to accident conditions.
- **Design Basis Accidents** are conditions for which a reactor facility is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits.
- **Design Extension Conditions** are postulated accident conditions that are less frequent than DBAs. DECAs are a subset of beyond-design-basis accidents (BDBA), and are therefore, not part of the design basis. DECAs are considered in the design process of the facility in accordance with best-estimate methodology DECAs can occur without core damage or with core damage where releases of radioactive material are reasonably contained and kept within acceptable limits.

BDBAs other than DECAs are accidents for which confinement of radioactive materials cannot be reasonably achieved. These are referred to as severe accidents and involve a catastrophic failure, core damage, and fission product release. A severe accident is generally considered to begin with the onset of core damage.

Representative DECAs with core damage are postulated to provide inputs for the design of the containment and of the safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECAs and represents a set of bounding cases that envelope other severe accidents with more limited degradation of the core.

These accidents scenarios are considered for practical elimination as described in Subsection 3.1.8.

Events are assigned to a plant state based on the expected frequency of the fault sequence, which includes a Postulated Initiating Event (PIE) and, in some cases, additional failures of mitigating functions. As described in CNSC REGDOC-2.5.2, Section 7.4 (Reference 3.1-1), PIEs are the events that lead to deviations from normal operation. PIEs originate from operating errors, equipment failures, or internal or external hazard of natural or human origin.

Frequency ranges for plant states are:

- AOO (greater than 1E-02 per reactor-year)
- DBA (1E-02 to 1E-05 per reactor-year)
- DEC (less than 1E-05 per reactor-year)

The design requirements of SSC are developed to ensure that the plant is capable of meeting applicable requirements for each plant state. This is demonstrated through safety analyses as described in Chapter 15.

The facility is operated, monitored, and maintained within safe operating configurations or is transitioned to a safe operating configuration in accordance with operating procedures that are consistent with the design. (See Chapter 13, Section 13.4 for details.)

Acceptance criteria are assigned to each plant state in the design, considering the principle that frequent fault sequences have only minor or no radiological consequences, and that any fault sequences that may result in severe consequences are of extremely low probability.

For normal operating modes, the OLCs serve as acceptance criteria as they are the set of limits and conditions within which the facility must be operated to ensure it is operated safely. OLCs are established as discussed in Chapter 16.

For each AOO and DBA fault sequence, acceptance criteria are defined and met to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These acceptance criteria are discussed and summarized in Chapter 15, Section 15.3.

For DEC fault sequences, the safety objectives are to prevent significant core damage, mitigate accident consequences, and protect containment integrity. These objectives are demonstrated in PSA by showing that the plant meets the established safety goals (described in Subsection 3.1.2.3). (PSA is described in detail in Chapter 15, Section 15.6.) Also, it is demonstrated that procedures and equipment put in place to handle accident management needs are effective in responding to DEC. This is accomplished through the operating procedures described in Chapter 13 and through complementary design features described in Chapter 15, Appendix 15B.

The general approach to defining the design basis for the BWRX-300 involves establishing the plant states described above, identifying the PIEs leading to a deviation from normal operation and categorizing mitigating functions based on their ability to prevent and mitigate the progression of events ensuring that the safety objectives are met.

### **3.1.4 Prevention and Mitigation of Accidents**

The design of the BWRX-300 includes provisions to prevent and to mitigate the consequences of accidents and to ensure that the likelihood that an accident will have harmful consequences is extremely low.

The primary means of preventing and mitigating the consequences of accidents is through the application of Defence-in-Depth (D-in-D). The application of D-in-D for the BWRX-300 design is described below in Subsection 3.1.6.

### **3.1.5 Fundamental Safety Functions**

The design of the BWRX-300 fulfills Fundamental Safety Functions (FSFs) at all plant states (defined in Section 3.1.3) which ensures the design meets the safety objectives consistent with CNSC REGDOC-2.5.2, Section 6.2 (Reference 3.1-1). The FSFs for the BWRX-300 are:

- Control of reactivity
- Removal of heat from the fuel (in the reactor, during fuel storage and handling, and including long-term heat removal)
- Confinement of radioactive materials, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental releases

The FSF prevent or mitigate radiological releases by ensuring the physical barriers to releases (fuel matrix, fuel cladding, Reactor Coolant Pressure Boundary (RCPB), and containment) remain effective. In addition to the protection of barriers, a means of monitoring the status of key plant parameters is provided for ensuring that the FSF are fulfilled. From this perspective, the monitoring function is treated as inherent to the design of the FSF. Other considerations for the monitoring function are as follows:

1. If a manual operator action plays a role in performing an FSF, the monitoring function of the equipment used to display key plant parameters that are necessary for the operator to perform the manual action successfully are also considered part of the FSF.
2. Certain monitoring functions allow the operator to confirm ongoing effectiveness of the FSFs during all plant states, to implement post-accident procedures, and to make decisions in support of emergency planning.
3. Post-Accident Monitoring (PAM) is important for operator decision making such as taking manual actions and implementing functions. Therefore, the designation, treatment and display of certain plant parameters or measurements as post-accident monitoring variables is a supporting design feature.
4. A minimum set of monitoring functions and display of parameters that do not support the operator actions are provided to support accident assessment.

Preservation of the FSFs is intrinsic to BWRX-300 Safety Strategy. A systematic approach is taken to identify the FSFs and those SSC necessary to fulfill the FSFs following a PIE. This systematic approach is detailed in the D-in-D discussion below.

### **3.1.6 Defence-in-Depth**

#### **3.1.6.1 BWRX-300 Defence-in-Depth Concept**

The implementation of D-in-D in the BWRX-300 design is the basis for the Safety Strategy for ensuring an adequate level of safety is achieved by the design.

The concept of D-in-D (consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1)) involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of a nuclear power plant, the undesirable outcome is the exposure of workers, the public or the environment to radioactivity exceeding levels determined to be safe.

There are two types of defensive layering considered:

1. Physical barriers in place to prevent the release of radioactivity: The fuel matrix, fuel cladding, RCPB, and containment. The integrity of one or more physical barriers must be maintained to prevent unacceptable releases.
2. A combination of active, passive, and inherent safety features used to minimize challenges to the physical barriers, to maintain the integrity of the barriers and, in case a barrier is breached, to ensure the integrity of the remaining barriers.

While the physical barriers themselves represent multiple layers of defence against radioactive releases, in the BWRX-300 D-in-D application, the physical barriers are not themselves referred to as “defense lines”. That term is reserved for the layers of defence comprising features, functions and practices that protect the integrity of the barriers. The D-in-D concept applied is largely focused on identifying and organizing features, functions, and practices into defense lines without explicit acknowledgment of the physical barriers. The fundamental purpose of the defense lines is to ensure the integrity of the physical barriers by applying multiple levels of protection.

The BWRX-300 D-in-D concept uses the FSFs described above to define the interface between the defense lines and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers remain effective.

### 3.1.6.2 Defense lines

Five Defense Lines (DLs) (or levels), DL1 through DL5, are adopted consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1) and IAEA SSR-2/1 (Reference 3.1-2). Figure 3.1-1 illustrates the defense lines as they correspond to the plant states.

The first defense line (DL1) does not include plant functions. It minimizes potential for PIEs to occur in the first place and minimizes potential for failures to occur in subsequent defense lines by assuring high quality and conservatism in design, construction, and operation. The second, third, and fourth defense lines (DL2, DL3, and DL4) comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The fifth defense line (DL5) involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs.

The defense lines include measures such as engineering and operational practices, plant features, and plant functions. These measures are incorporated such that:

- The normal operation of the plant is monitored and controlled such that PIEs that lead to AOOs can be mitigated before evolving into DBAs
- The consequences are limited if a DBA does develop
- Multiple defense lines are capable of independently performing the FSFs. While this means that more than one DL is capable of independently performing the FSFs for D-in-D, DL independence from all other DLs is based on how specific DLs are credited for specific fault sequences.

Table 3.1-1 provides a high-level description of the objective, and the design means and operational means for supporting the defense lines. The following is a brief description of each of the defense lines.

#### Defense Line 1 (DL1)

The purpose of the first level of defence is to prevent deviations from normal operation and the failure of important SSC. This is achieved through the quality measures taken to minimize potential for failures and for initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defence. These quality measures cover the design, construction, inspections, operation, use of operational experience, periodic safety reviews, and maintenance, and testing of the plant.

DL1 measures may support the basis for assumptions made in safety analyses. For example, the use of a high-quality design process and stringent equipment qualification for the most important components support the assumption that only a single failure is considered in the Conservative Deterministic Safety Analysis discussed in Chapter 15, Subsection 15.2.1.

Examples of DL1 measures include:

- The clear definition of normal and abnormal operating conditions
- Maintenance and implementation of a quality assurance program consistent with nuclear regulations and industry standards
- Application of appropriate industry standards to the design of SSC
- Adequate design margins
- Robust design processes including design verifications
- Comprehensive testing programs

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Provisions for adequate time for operators to respond to events and appropriate human-machine interfaces, including operator aids, to reduce the burden on the operators
- Deterministic safety analyses including appropriate conservatism, supplemented by Probabilistic Safety Analysis to produce risk insights
- Categorization and qualification of SSC according to their safety significance
- Operational Limits and conditions
- Application of lessons learned through operating experience

**Defense Line 2 (DL2)**

The purpose of the second level of defence is to detect and control deviations from normal operational states to prevent AOOs from escalating to accident conditions. Functions that normally operate to maintain key reactor parameters (e.g., pressure, reactor level, and reactivity) within normal ranges are part of DL2.

Examples of DL2 measures include:

- Anticipatory plant trips
- Maintain target power
- Maintain target level
- Maintain target pressure
- Control Rod Block

**Defense Line 3 (DL3)**

For the third level of defence, it is assumed that, although very unlikely, the escalation of certain AOO or DBA PIEs might not be controlled at a preceding level and that an accident could develop. In the design of the plant, such accidents are postulated to occur. DL3 contains plant functions that act to mitigate a PIE by preventing fuel damage, when possible, which assures the integrity of the release barriers are maintained, and the plant is maintained in a safe state until normal operations are resumed.

The systems and equipment involved in performance of DL3 functions are designed for high reliability. Examples include eliminating the need for active support systems such as power supplies, ventilation, or cooling water, and minimizing the need for active control functions such as pumps and actively controlled valves.

The DL3 functions and equipment performing those functions are subject to functional and design requirements derived from the Conservative Deterministic Safety Analysis as described in Chapter 15, Subsection 15.2.1.

Examples of DL3 measures include:

- Reactor Scram
- Isolation Condenser Initiation
- Main Steam isolation
- Containment Isolation
- Reactor Pressure Vessel (RPV) Isolation



#### **Defense Line 4 (DL4)**

The purpose of the fourth level of defence is to mitigate DEC's.

For the BWRX-300, DL4 is comprised of two subsets of functions that are designated as DL4a and DL4b functions. DL4a functions mitigate DEC's that occur without core damage. DEC's progressing to core damage are mitigated by DL4b functions.

##### **DL4a**

DL4a functions are those that place and maintain the plant in a safe state in scenarios involving:

- DBAs sequences combined with multiple failures that prevent the DL3 SSC from performing their intended function (i.e., Common Cause Failure (CCF) which is a failure of two or more SSC due to a single specific event or cause.)
- DEC PIEs considered as credible events that may involve multiple failures causing the loss of a FSF to be fulfilled as part of normal operation

Examples of DL4a measures include:

- Diverse means of achieving the FSFs that are independent of and diverse from the SSC carrying out the DL3 functions that are presumed to have failed.
- Scrams initiated by the Diverse Protection System

##### **DL4b**

DL4b includes:

- Functions provided in scenarios leading to core damage to limit the radiological releases in case of core damage and are aimed at maintaining the containment functions for extreme events, multiple events, or multiple failures that defeat DL2, DL3, and DL4a.
- Functions provided to mitigate the effects from a damaged core and to preserve the FSF of confinement of radioactive material while limiting radioactive releases to acceptable levels.
- Safety features designated for DEC's with core damage may, if practicable and available, also be used for preventing or minimizing significant core damage if it can be demonstrated that such use will not undermine the ability of these systems to perform their primary functions if conditions evolve into a severe accident.

Examples of DL4b measures include:

- DL4b measures carried out by complementary design features such as diverse and flexible equipment and portable components such as, portable uninterruptible power supplies and portable pumps
- Containment venting and overpressure protection
- Boron injection

A list of complementary design features is provided in Chapter 15, Appendix 15B.

#### **Defense Line 5 (DL5)**

The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accidents.

DL5 includes emergency preparedness measures to cope with potential unacceptable releases in case the first four defense lines are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

Examples of DL5 measures:

- Severe accident management procedures
- Emergency response procedures and equipment (peripheral systems such as meteorological monitoring)
- On/off-site emergency response facilities, and certain communication systems may play a role in DL5. Chapter 19 discusses emergency response arrangements such as procedures and facilities. Communication systems are discussed in Chapter 9A, Section 9A.9.1. (Note that these measures may be initiated earlier in an event prior to progression to a severe accident)

### **3.1.6.3 Defense Line Independence**

The BWRX-300 design incorporates independence in the application of D-in-D. Defense lines that mitigate the same event are independent as far as is practicable to avoid the failure of one level reducing the effectiveness of other levels. Some examples include:

1. Among DL2, DL3 and DL4a, at least one defense line can mitigate a PIE caused by or concurrent with a CCF in another defense line, with the mitigation means being independent from the effects of the initiating CCF.
2. All PIEs with a frequency greater than  $1\text{E-}05$  caused by a single failure can be mitigated by DL3 and independently by DL2, DL4a, or a combination of DL2 and DL4a functions that are unaffected by the PIE. To the extent practicable, DL3 functions are independent and diverse from those in DL2 and from those in DL4a. This is because DL3 functions provide a backup to DL2 functions, and DL4a functions provide a backup to DL3 functions but DL4a functions are not needed to provide a direct backup to DL2 functions to maintain D-in-D for the same event.
3. The DL4b functions intended for mitigating DECIs are functionally and physically separated from the systems intended for other DL functions.
4. DL4b features specifically designed to mitigate the consequences of accidents with core damage are independent from systems used in normal operation or used to mitigate AOOs as far as is practicable and with exceptions justified.
5. Exceptions to rules of independence are described, assessed, and justified. If equipment supports functions in more than one defense line, there is an increased focus on their reliability in the application of DL1 compared to a design feature credited in only one defense line.

### **3.1.6.4 Safety Strategy Process for Implementing Defence-in-Depth**

The BWRX-300 Safety Strategy implements the D-in-D concept into the design through evaluations and analyses as shown in Figure 3.1-2. These include:

- Hazard Evaluations
- Fault Evaluation
- Deterministic Safety Analyses
- PSA

The elements of Figure 3.1-2 are briefly described below.

#### **3.1.6.4.1 Hazard Evaluations**

The first step is to identify PIEs using a systematic methodology considering both direct and indirect events through hazard evaluations. The BWRX-300 Safety Strategy includes the following four types of hazard evaluations which are summarized in Chapter 15, Subsection 15.1.3:

- Functional Failure Hazard Evaluation – assessment of failures of SSC
- External Hazard Evaluation - assessment of external events such as earthquakes or aircraft crashes that have the potential to impact plant safety
- Internal Hazard Evaluation – assessment of hazards originating within the facility such as missiles from rotating equipment, fires, collapse of structures
- Human Operation Hazard Evaluation – human errors which could reasonably be expected to occur based on industry operating experience

The output of the four hazard evaluations are the potential PIEs for consideration in the Fault Evaluation.

#### **3.1.6.4.2 Fault Evaluation**

The Fault Evaluation process evaluates the PIEs determined as a result of the hazard analyses. PIEs are selected and organized along with fault sequences. As used herein, a fault is essentially a failure or a hazard and could be the initiator for or result from a PIE. A PIE is an event that initiates a fault sequence. A fault sequence consists of a PIE, and responses by mitigation functions (including both failed responses and successful responses). This is consistent with the description of event combinations per CNSC REGDOC-2.4.1, Section 4.2.2.5 (Reference 3.1-5).

The Fault Evaluation establishes traceability between the plant design and the safety analysis bases. The Fault Evaluation process including the selection and categorization of PIEs and fault sequences for deterministic safety analysis is described in Chapter 15, Section 15.2.

#### **3.1.6.4.3 Deterministic Safety Analyses**

The objective of deterministic safety analysis for nuclear power plants is to confirm that:

- FSFs can be performed
- SSC performing the FSF are designed with adequate margins
- physical barriers to radioactive release maintain their integrity as required

Deterministic safety analysis is supplemented by insights obtained from fabrication, testing, inspection, operating experience, and PSA. It demonstrates that the source term and the potential radiological consequences of different plant states are acceptable. It also demonstrates that the possibility of certain conditions arising that could lead to an early or a large radioactive release can be considered as 'practically eliminated'.

The output of the Fault Evaluation process which includes the selection of PIEs and fault sequences organized by frequency are analyzed in deterministic safety analysis. Chapter 15, Subsection 15.2.1, provides more detail on the deterministic safety analysis process.

#### **3.1.6.4.4 Probabilistic Safety Analyses**

PSA are performed to understand the overall risk presented by the facility and to allow comparisons to be made against safety goals (defined in Section 3.1.2.3) They also provide

essential understanding of strengths and weaknesses of a design with complex systems and interdependencies. They are used for evaluating complementary design feature concepts or changes in operating conditions and have many other applications to enhance safety decision

To supplement quantitative PSA results, a severe accident analysis is performed to understand the complex physical phenomena associated with a reactor core damage scenario. This analysis supports confirmation that the radioactive release sequences modeled in the Level 2 PSA adequately reflect associated phenomena.

Severe accident analyses are used to complement the design deterministic safety and PSA in situations where the consequence is large, even if the calculated risks are low and/or the deterministic safety analysis provides a robust demonstration of fault tolerance. The severe accident analysis is not considered standalone piece of analysis deriving scenarios from first principles, but instead builds upon other types of analysis to create an overall safety case that is adequate in its coverage.

Detailed discussion of PSA and Severe Accident Analysis is provided in Chapter 15, Section 15.6.

### **3.1.7 Application of General Design Requirements and Technical Acceptance Criteria**

#### **3.1.7.1 Deterministic Design Principles in Codes & Standards**

A fundamental aspect of the BWRX-300 Safety Strategy is that the overall plant design applies good engineering practices for design, construction, operation, maintenance, and testing which relates to conformance to regulatory requirements, as well as industry codes and standards and norms for achieving high dependability in performance.

Engineering design rules are established and applied, as appropriate by the specific design discipline based on relevant codes, standards, and proven engineering practices.

Because codes or standards for the different design disciplines (e.g., mechanical, civil, and electrical) are not always based on compatible safety criteria, consistent acceptance criteria are established, and good engineering practices are used, to provide consistency in the application of selected codes and standards in design. Analyses and evaluation of the codes and standards to be applied in the design, fabrication and construction of the plant is performed. The results of this analysis and evaluation are documented as part of the management system.

The plant architecture and systems design specifications demonstrate that the plant and the SSC are designed, implemented, constructed, installed, operated, and maintained safely with respect to their application and maintenance of these guiding fundamental design principles that follow. Additionally, changes are performed using the same guiding fundamental design principles, using the same or better methods and processes to avoid compromising safety.

#### **3.1.7.2 Minimize Probability of Failure Structures, Systems, and Components**

The probability of failure of systems and equipment is minimized through a design which provides predictable and repeatable performance of the FSFs. This is achieved by deploying highly reliable and dependable SSC.

DL3 systems and equipment are designed to fail to a safe state or to a known, defined state to ensure safety is not jeopardized. Thus, reactor trip systems fail to the safe state, but engineered safety features systems may fail-safe or are non-actuated (e.g., isolation condenser cooling function). Fail-safe design is achieved through systematic identification of failure modes through Failure Modes and Effects Analyses (FMEA).

Systems are required to be testable to provide assurance of continued operability and availability when required. System maintainability is a fundamental aspect of the design, extending down to software by ensuring documented, well-designed, understandable code.

Chapter 13 describes how fitness for service is addressed in established programs that include: Reliability, Maintenance, Aging Management, Chemistry Control, Periodic and In-Service Inspections. Programmatic requirements addressing fitness for service span the full life cycle of the facility beginning with inclusion in facility design decision making.

#### **3.1.7.3 Independence**

The most plausible reason for the failure of FSFs is the occurrence of dependent failures. Dependent failures are identified, and where practicable, measures are implemented in design, construction, and operation to eliminate the dependencies or reduce their potential effect. The application of independence is used in the Safety Strategy to enhance reliability and reduce potential for dependent failures. Independence is an essential aspect of effectiveness in the implementation of D-in-D.

The determination of independence of SSC required to mitigate the consequences of a single or a likely combination of internal or external hazards on the plant is conducted through the Fault Evaluation introduced in Section 3.1.6.4.2 and described in more detail in Chapter 15, Section 15.2 and confirmed via the PSA in Chapter 15, Section 15.6.

The PSA is also used to confirm the adequacy of the independence measures.

Independence is achieved by addressing the main causes of CCFs: functional, spatial, inherent, and human error dependencies as discussed in Subsection 3.1.7.5.

#### **3.1.7.4 Diversity**

Diversity is the provision of dissimilar means of achieving the same objective. Diversity involves the use of design features which differ in the physical means of achieving a specific objective or use of different equipment made by different manufacturers. Diversity is achieved by incorporating different attributes into the systems or components. Such attributes could be different principles of operation, different physical variables, different conditions of operation, or production by different manufacturers, for example. It is necessary to ensure that the diversity attribute achieves the desired increase in reliability in the as-built design. For example, to reduce the potential for CCFs the designer should examine the application of diversity for any similarity in materials, components and manufacturing processes, or subtle similarities in operating principles or common support features. If diverse systems or components are used, there is a consideration that reasonable assurance that such additions are of overall benefit, including consideration of the associated disadvantages such as the increased operational complication, additional maintenance and test procedures, and the potential for lower reliability.

Diversity is considered for digital equipment and active mechanical/electrical equipment. Diversity is not included for passive equipment such as pipes and tanks. Diversity is a DL1 provision used to strengthen subsequent defense lines.

#### **3.1.7.5 Separation**

Functional isolation is used to reduce the likelihood of adverse interactions between equipment and components resulting from normal or abnormal operation or failure of any component in the systems. For example, in a power supply, functional isolation is commonly achieved using fuses and circuit breakers.

Separation supports defense line function independence discussed in Subsection 3.1.6.3. System layout and design uses physical separation to increase assurance that independence will be achieved, to preclude certain CCFs.

- Physical separation includes separation by geometry (such as distance or orientation); barriers; or a combination of these. The choice of the means of separation will depend on the PIEs considered in the design basis, such as the effects of fire, chemical explosion, aircraft crash, missile impact, flooding, extreme temperature, or humidity.
- In a redundant system and despite diverse provisions, the threat of CCFs from hazards such as fire may be reduced by system segregation. Segregation is the separation of components by distance or physical barriers. An example is the use of fire barriers to indicate individual fire zones, which may also serve as barriers to other hazards.
- Plant barriers that provide protection against certain faults or hazards are assessed to ensure that the barriers remain operable and accessible in the event of those faults or hazards occurring. This is particularly important where SSC that perform defense line functions are co-located with other plant equipment that do not.

### **3.1.7.6 Redundancy**

Redundancy is the provision of more than the minimum number of nominally identical equipment items required to perform a specific safety function. Such redundant provisions allow a safety function to be satisfied when one or more systems or components (but not all) are unavailable, due to a variety of unspecified potential failure mechanisms or maintenance (e.g., identified faults or hazards). Redundancy enables failure or unavailability of at least one set of systems or components without loss of the function. For example, three or four pumps may be provided for a particular function when any two would be capable of carrying it out. For the purposes of redundancy, identical or diverse components may be used.

The application of independence, diversity, separation, and redundancy in the design is described in each system design description.

### **3.1.7.7 Single Failure Criterion**

The BWRX-300 design addresses the single failure criterion through design and safety analyses to ensure reliability of DL3 functions. Consistent with CNSC REGDOC-2.5.2, Section 7.6.2, each safety group (DL3 function) is assessed for capability in fulfilling its required function even if a failure of a single component occurs within this group.

A single failure is one which results in the loss of capability of a single system or component to perform its intended DL3 function(s), and any consequential failure(s) which result from it.

For the BWRX-300, the single failure criterion is considered in two ways:

1. As a design attribute that is typically achieved through redundancy in the system architecture of the SSC carrying out DL3 functions. This involves a systematic search for potential single failure points and their effects on prescribed missions (i.e., FMEA).
2. As an assumption made in the conservative deterministic safety analysis, in addition to the PIE and any additional failures, all identifiable undetectable faults are included to demonstrate a high degree of confidence that acceptance criteria will be met.

During the design process, systems that are designed to carry out a DL3 function must be capable of carrying out their mission despite the failure of any single component within the system or in an associated system that supports its operation. Design measures for ensuring high reliability

of SSC carrying out DL3 functions include incorporating, independence, diversity, and redundancy, and also through the incorporation of passive and fail-safe features.

The PSA is used for identifying single failures for consideration in the deterministic safety analysis and is also a complementary means of demonstrating the insensitivity to single failures.

### **3.1.7.8 Common Cause Failures**

#### ***3.1.7.8.1 Background Information and General Approach***

CCFs are functional failures of multiple components due to a single specific event or cause. Such failures may affect several different safety class components simultaneously or may affect multiple components of the same type at the same time.

The event or cause of CCFs may be a design deficiency, a manufacturing deficiency, an operating or maintenance error, a natural phenomenon, a human induced event or an unintended cascading effect from any other operation or failure within the plant. Appropriate measures to minimize the effects of CCFs, such as the application of redundancy, diversity, and independence, are taken as far as practicable in the design.

Multiple failures can occur due to common weaknesses or dependencies shared by components. Such failures can cause failure of all redundant components in a single protection system or failure of components in more than one system. Dependent failures can considerably reduce the reliability of the protection systems relative to that expected from consideration of random failure mechanisms occurring in isolation. Identification of dependent failures is assessment by Functional Failure Hazards Evaluations.

The main types of failure dependencies that can cause a potential loss of safety function are as follows:

- **Functional Dependencies**, which arise from shared or common functional features such as a common electrical power source, a common cooling water system or a shared process fluid.
- **Spatial Dependencies**, which arise from physical features shared by components located in a common location such as common radiation or chemical conditions, a common environment and common support structures, and vulnerability to leaks of dangerous fluids (high temperature, corrosive or toxic).
- **Inherent Dependencies**, which arise from shared characteristics such as a common principle of operation or technical embodiment and a common failure mechanism such as mechanical overload or overpressure.
- **Human Error Related Dependencies**, which arise from human errors affecting some shared or common human process such as human error in design or manufacture, or operating staff error during operation and maintenance.

The general protective approach used for addressing postulated vulnerabilities to CCFs is diversity in the design. Dissimilarities in technology, function, implementation, and so forth, can mitigate the potential for common faults. The diversity approach to ensuring safety uses different (e.g., dissimilar) means to accomplish the same or equivalent function to compensate for a CCF that disables one or more levels of defence. Diversity is complementary to the principle of defence-in-depth, and it increases the chances that a defense line function will be available when needed. Different defense lines that mitigate the same event are diverse from each other to the extent practicable.

Another means of protecting against CCF is through feedback from operating experience that could identify weaknesses in the design, construction, operation and testing of equipment. In addition, conducting periodic inspection, surveillance, and testing provides opportunities to detect degradation or common causes before failures of SSC. Quality assurance and quality control measures applied to SSC commensurate with their importance help reduce preclude potential CCFs.

#### ***3.1.7.8.2 Common Cause Failures of Digital Instrumentation and Control Software***

The BWRX-300 approach to assessment of CCF of Digital Instrumentation and Control (I&C) software is through a consequence-based approach.

Even when functional dependencies are addressed through rigorous design and application of codes and standards, operating experience shows that software CCFs occur. Validating assumptions and modeling of software CCF modes can be challenging due to uncertainty as each Digital I&C system is unique, and extrapolation of failure data from one system to another may not be meaningful making the identification of failure scenarios difficult. Analyzing each postulated CCF scenario is not practicable; therefore, using a consequence-based approach can limit the number of CCF scenarios is considered. This approach considers the radiological or dose consequences that could result due to CCFs in the software.

#### ***3.1.7.8.3 Defense Line Approach to Common Cause Failure***

A multi-pronged approach and the systematic integration of CCFs in defense line functions, both as PIEs and as failures affecting fault sequence mitigation, are applied in deterministic safety analyses for prevention and mitigation in the D-in-D approach. Examples include:

1. DL3 systems and functions are designed and rigorously qualified to be resistant to the effects of environments that could cause common failures, including DBA environments.
2. For internal and external events resulting in DECAs, the design includes independent and diverse system functions to cope with the effects of common cause failure (e.g., DL4a).
3. Diverse accident monitoring instrumentation for severe accident management (e.g., DL4b) is provided.

The defence-in-depth approach is designed to include analyses of a reasonable set of CCF scenarios to provide assurance that the plant is protected against CCF phenomena. This approach is implemented using a set of CCF application guidelines to define the CCF modes that are included, how the failure modes are applied, and which assumptions can be made regarding equipment operability.

#### **3.1.7.9 Other Approaches for Ensuring Safety**

In addition to the design principles discussed above, the BWRX-300 design incorporates the following approaches to ensure safety.

##### ***3.1.7.9.1 Simplicity in Design***

An implicit approach to reliability is to deploy the design with minimal complexity, with the knowledge that complexity may be required to enhance reliability or reduce the potential for human error. Where complexity is required (e.g., self-diagnostics, redundancy within the equipment in a single division), the complexity is documented and justified as necessary and appropriate for enhancing reliability, surveillance, calibration, and other required system or equipment attributes. There are tradeoffs in complexity, such as increasing the complexity by designing the system to reduce the human actions necessary for surveillance which also decreases the potential for human error, which enhances system reliability.



The BWRX-300 is specifically designed to enhance safety through simplification and reducing its dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems (these are passive features as discussed below). These safety enhancements, in combination with its reduction in scale and complexity including a reduction in total number of active SSC, simplifies operations and maintenance. Some of the simplified design features are described in Chapter 1.

#### **3.1.7.9.2 *Passive Safety Features***

The design of the BWRX-300 uses passive functions that do not require external sources of power or operator actions. DL3 functions are passive to the extent that is practicable and, therefore, have significantly less reliance on supporting systems or operator actions.

Examples of the BWRX-300 passive design features include:

1. Safety Class 1 batteries are capable of powering loads for a minimum of 72 hours. The design ensures that plant safety is maintained even after battery depletion.
2. BWRX-300 utilizes natural circulation and passive natural circulation for fuel cooling and passive containment heat removal. The plant is designed with the capability to cope with decay heat for seven days using only installed systems with no reliance on significant operator actions or external resources.

The mitigation of loss-of-coolant accidents is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, reduced number, and size of RPV nozzles as compared to predecessor designs, and elimination of fluid system nozzles located below a level well above the top of active fuel to conserve inventory. The relatively large reactor pressure volume of the relatively tall chimney region provides a substantial reservoir of water above the core. This ensures the core remains covered following fault sequences involving feedwater flow interruptions or loss-of-coolant accidents without the need for active components (such as pumps). Additionally, the RPV is equipped with isolation valves attached directly to the reactor vessel for large bore piping systems to preserve reactor coolant inventory ensuring that adequate core cooling is maintained.

The application of these design concepts is described in each system design description.

#### **3.1.7.10 Technical Acceptance Criteria**

To meet the radiological acceptance criteria, derived accepted criteria are defined for the fuel pellet, fuel cladding, RCPB and containment. Deterministic safety analyses are performed to demonstrate that these criteria have been met. A description of acceptance criteria is provided in Chapter 15, Section 15.3. Details of the deterministic safety analysis are presented in Chapter 15 Section 15.3. Table 15.3-1 for AOOs and 15.3-2 for DBAs.

#### **3.1.8 Practical Elimination**

Consistent with CNSC REGDOC-2.5.2 Section 7.3.4 (Reference 3.1-1) and IAEA SSR-2/1(Reference 3.1-5), the BWRX-300 design is such that fault sequences that could lead to an early or large radioactive release are practically eliminated.

The definition of early and large radioactive release (from IAEA SSR-2/1) (Reference 3.1-5) in this context are:

1. An early radioactive release is a release of radioactive material for which off-site protective actions would be necessary but would be unlikely to be fully effective in due time.

2. A large radioactive release is a release of radioactive material for which off-site protective actions that are limited in terms of lengths of time and areas of application would be insufficient for the protection of people and of the environment.

Fault sequences with early or large releases could be considered to have been practically eliminated if either of the following apply:

- It is physically impossible for the accident sequence to occur.
- The fault sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

Practical elimination is considered to refer only to those fault sequences leading to or involving core damage (e.g., a severe accident) for which the confinement of radioactive materials cannot be reasonably achieved.

The aim of the practical elimination concept is to reinforce D-in-D by focused analysis of those conditions having the potential for early radioactive release or a large radioactive release.

The justification of practical elimination preferably relies on a demonstration of physical impossibility for the accident sequence to occur. If this is not achievable, a demonstration of an extremely low likelihood of occurrence with a high level of confidence is provided. Sufficiently robust arguments and evidence are used to demonstrate the reliability of the lines of defence. If additional features are identified that prevent accidents or mitigation accident consequences, these features are considered for implementation as far as practicable.

The set of individual fault sequences that might lead to an early radioactive release or a large radioactive release are grouped to form a limited number of bounding cases or type of accident conditions.

Severe accident phenomena based on operating experience with predecessor advanced light water reactors serve as a starting point for consideration for practical elimination. Analyses demonstrating practical elimination are described in Chapter 15, Appendix 15A.

### **3.1.9 Safety Margins and Avoidance of Cliff-Edge Effects**

A cliff-edge effect is described as a small change of conditions that may lead to a significant increase in the severity of consequences per CNSC REGDOC-3.6 (Reference 3.1-7).

In the BWRX-300 Safety Strategy, the principle of multiple physical barriers to the release of radioactive material and protection of those barriers is incorporated in the design as a DL1 measure. Margins are incorporated into the design of the physical barriers to demonstrate their capability in postulated scenarios that are more severe (by a small amount) than those in the design basis without incurring cliff-edge effects.

Conservative safety margins and sensitivity analyses are applied in safety analyses to account for assumptions and uncertainties. Additional details on the application of safety margins in Deterministic Safety Analysis are described in Chapter 15, Subsection 15.5.1.1. As part of the PSA, sensitivity and uncertainty analysis is conducted to demonstrate consideration of potential cliff-edge effects. (See Chapter 15, Subsection 15.6.1).

### **3.1.10 Design Approaches for the Reactor Core and for Fuel Storage**

#### **3.1.10.1 Design Approach for Reactor Core**

The reactor core is designed to maintain the integrity of the fuel and the fuel cladding. The fundamental safety functions of control of reactivity, removal of heat from the reactor and fuel, and confinement of radioactive materials are inherent design features for the reactor core.

The reactor core, the fuel, and fuel assemblies, including fuel channels and control blades, are designed such that the reactor can be shut down, cooled, and held subcritical with adequate margin in operational states, DBAs, and DECAs. Reactivity control ensures shutdown margin for shutdown states and any credible changes in core configuration. The design ensures that the fission chain reaction is controlled during operational states. The design limits positive reactivity through inherent neutronic and thermal-hydraulic characteristics, means of shutdown, and control to protect the reactor pressure boundary and prevent fuel damage.

The reactor core (including associated structures and cooling systems) is designed to withstand static and dynamic loading and vibration, to be compatible with expected chemicals, and to meet thermal material and radiation damage limits.

The reactor core design also provides for certain operator actions in accident scenarios to maintain the reactor in a shutdown condition, such as actions that might be addressed in emergency operating procedures or severe accident management guidelines.

### **3.1.10.2 Design Approach for Fuel Handling and Storage**

The design of fuel handling and storage systems is consistent with the D-in-D approach applied to the reactor core with slightly different fundamental safety functions.

The design approach is to identify fuel handling and storage SSC that are necessary to fulfill the following fundamental safety functions for all plant states:

- Maintaining subcriticality of the fuel
- Removal of the decay heat from irradiated fuel
- Confinement of radioactive material, shielding against radiation as well as limitation of accidental radioactive releases

The Safety Strategy principle for fuel handling and storage is to leverage design and safety features in relation to fuel handling and storage that have been proven either in predecessor BWR applications or are based on operating experience.

Subcriticality is maintained by preventing criticality through use of geometrically safe configurations. The design of fuel storage systems considers the use of physical means or physical processes to increase subcriticality margins in normal operation to avoid reaching criticality during PIEs, including those PIEs arising from the effects of internal hazards and external hazards.

Fuel handling and storage systems are designed to maintain adequate fuel cooling capabilities for irradiated fuel ensuring that the fuel cladding temperature limits and/or the coolant temperature limits, as defined for operational states and accident conditions, are not exceeded.

The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed:

- With a capability to permit appropriate periodic inspection and testing of components safety features,
- With suitable shielding for radiation protection,
- With appropriate containment, confinement, and filtering systems,

- With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and
- To prevent significant reduction in fuel storage coolant inventory under accident conditions.

Appropriate systems are provided in fuel storage and radioactive waste systems and associated handling areas:

- To detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and
- To initiate appropriate safety actions

Refer to Chapter 9A, Section 9A.1 for a detailed description of the Fuel Handling and Storage Systems.

### **3.1.11 Considerations of Interactions Between Multiple Units**

Operating experience has demonstrated that interactions or shared equipment between multiple units can cause problems for the plant and for personnel. Lessons learned include:

- Significant interactions between multiple co-located radiological sources (e.g., reactor units, spent fuel pools, or dry fuel storage facilities) could result due to concurrent or consequential initiators.
- The timing of concurrent accident sequences involving multiple radiological sources on a site can challenge shared SSC, as well as resources available for severe accident management and emergency response to the event.

Site evaluations would address multiple reactors or other co-located facilities and determine if these need to be treated as external hazards (e.g., external radiation sources) in the design of the BWRX-300. See Chapter 2, Subsection 2.2.5 for more details.

Each BWRX-300 unit would have its own safety class systems and its own safety features for DEC's.

If multiple units are to be co-located, emergency planning and design and safety analyses, including consideration of CCFs in similarly design units, would demonstrate that sharing resources of equipment and personnel, including temporary equipment and emergency response personnel, would not be detrimental to plant operation, fuel storage, emergency planning, or accident management.

### **3.1.12 Design Considerations for Aging Management**

Aging of SSC is considered in the basic assumptions and in the input data to the safety, thermohydraulic and stress analyses. All system and component design specifications reference design requirements on aging, including those in the applicable codes and standards.

Aging and equipment qualification considerations are important aspects, complementary to each other in plant design. Equipment qualification is discussed in Section 3.9.

In designing components, system designers consider aging mechanisms and their effects on the safety, reliability, and performance of SSC for those that are well known and understood. Additionally, system designers collect information from operations feedback, research and development, vendor recommendations, maintenance and operating manuals, and expert insight, and make design decisions based upon shared knowledge. For BWRX-300 there exists significant operating experience and insights regarding individual degradation mechanisms that have been considered in the aging management programs. For example, the United States

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclear Regulatory Commission has developed a consistent approach to aging management in connection with licence renewal for operating plants.

Known aging phenomena are quantified and considered in the design of SSC. The design includes the effects of wear and all other known age-related degradation to ensure that safety and performance are maintained for the duration of their lifetime. If the component lifetime extends to the plant service life, as is the case for passive non-replaceable components, the design considers all normal and transitory operating conditions, including testing stressors, maintenance interventions and the consequences of plant and system outages. Analyzed DBAs are considered as part of the operating life and hence part of the design calculations.

In general, margins consist of design margins, operational margins, and safety margins. They account for uncertainties, assumptions, instrument feedback tolerances and ranges, unexpected transitory peaks, contingencies, and operating flexibility. Margins are mainly set to minimize the probability of component failure. Only the unquantifiable aging effects are included in the margin estimates.

Design documents include as a minimum, the following aging management topics:

1. A recommended strategy for aging management and prerequisites for its implementation.
2. Identification of safety class SSC in the plant that could be affected by aging.
3. Proposals for appropriate materials monitoring and sampling programs, where aging may affect the capability of critical SSC to perform their functions throughout the lifetime of the plant.
4. Appropriate consideration of operating experience with respect to aging.
5. Recommendations for aging management for safety class SSC (concrete structures, mechanical components, electrical and instrumentation and control components, cables, etc.) and measures to monitor and mitigate their degradation.
6. Equipment qualification requirements of safety class SSC.
7. General principles stating how the environment of structures, systems, and components are to be maintained within specified service conditions (location of ventilation, insulation of hot SSC, radiation shielding, damping of vibrations, submerged conditions and water chemistry, selection of cable routes, etc.).

### **3.1.13 References**

- 3.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.1-2 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.1-3 IAEA Safety Standards Series No. SF-1, "Fundamental Safety Principles," International Atomic Energy Agency.
- 3.1-4 Government of Canada SOR/2000-203, "Radiation Protection Regulations,"
- 3.1-5 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 3.1-6 IAEA TECDOC-1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.1-7 CNSC Regulatory Document REGDOC-3.6, "Glossary of CNSC Terminology."

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.1-8 CNSC Regulatory Document REGDOC-2.4.2, "Safety Analysis – Probabilistic Safety Assessment (PSA) for Nuclear Power Plants."
- 3.1-9 IEC 60880, "Nuclear power plants – Instrumentation and control systems important to safety – Software aspects for computer-based systems performing category A functions," International Electrotechnical Commission.

**Table 3.1-1: Identification of Defence Levels**

<b>Level of Defence/DL</b>	<b>Objective</b>	<b>Design Means</b>	<b>Operational Means</b>
Level 1/DL1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures
Level 2/DL2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features (Safety Category 3)	Abnormal operating procedures/emergency operating procedures
Level 3/DL3	Control of design basis accidents	Engineered safety features (Safety Category 1)	Emergency operating procedures
Level 4a/DL4a	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core damage (Safety Category 2)	Emergency operating procedures
Level 4b/DL4b	Control of design extension conditions to prevent or mitigate the consequences of severe accidents	Safety features for design extension conditions with core damage (Safety Category 3)	Complementary emergency operating procedures/severe accident management guidelines
Level 5/DL5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans

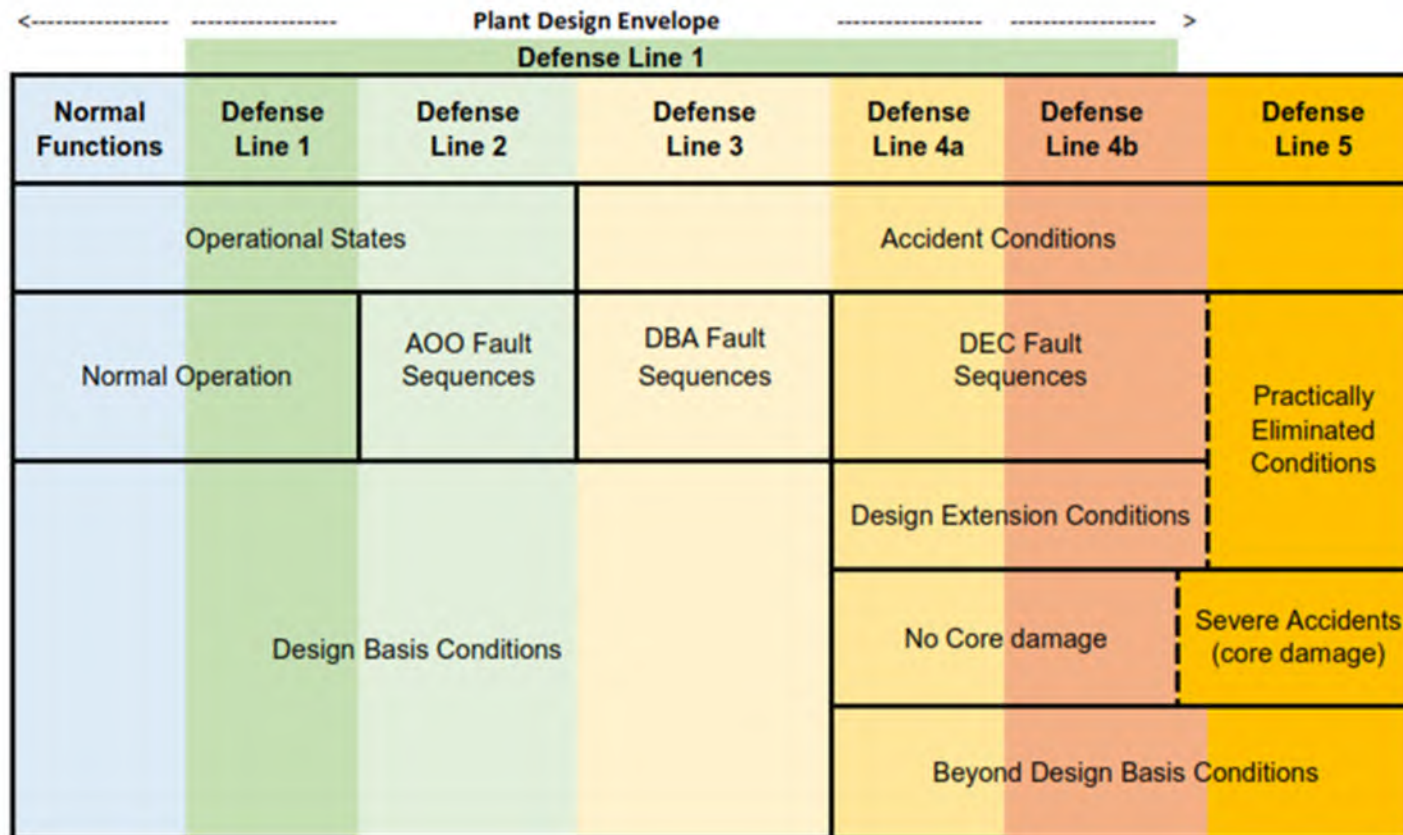


Figure 3.1-1: Defence-in-Depth – Plant States and Defense lines



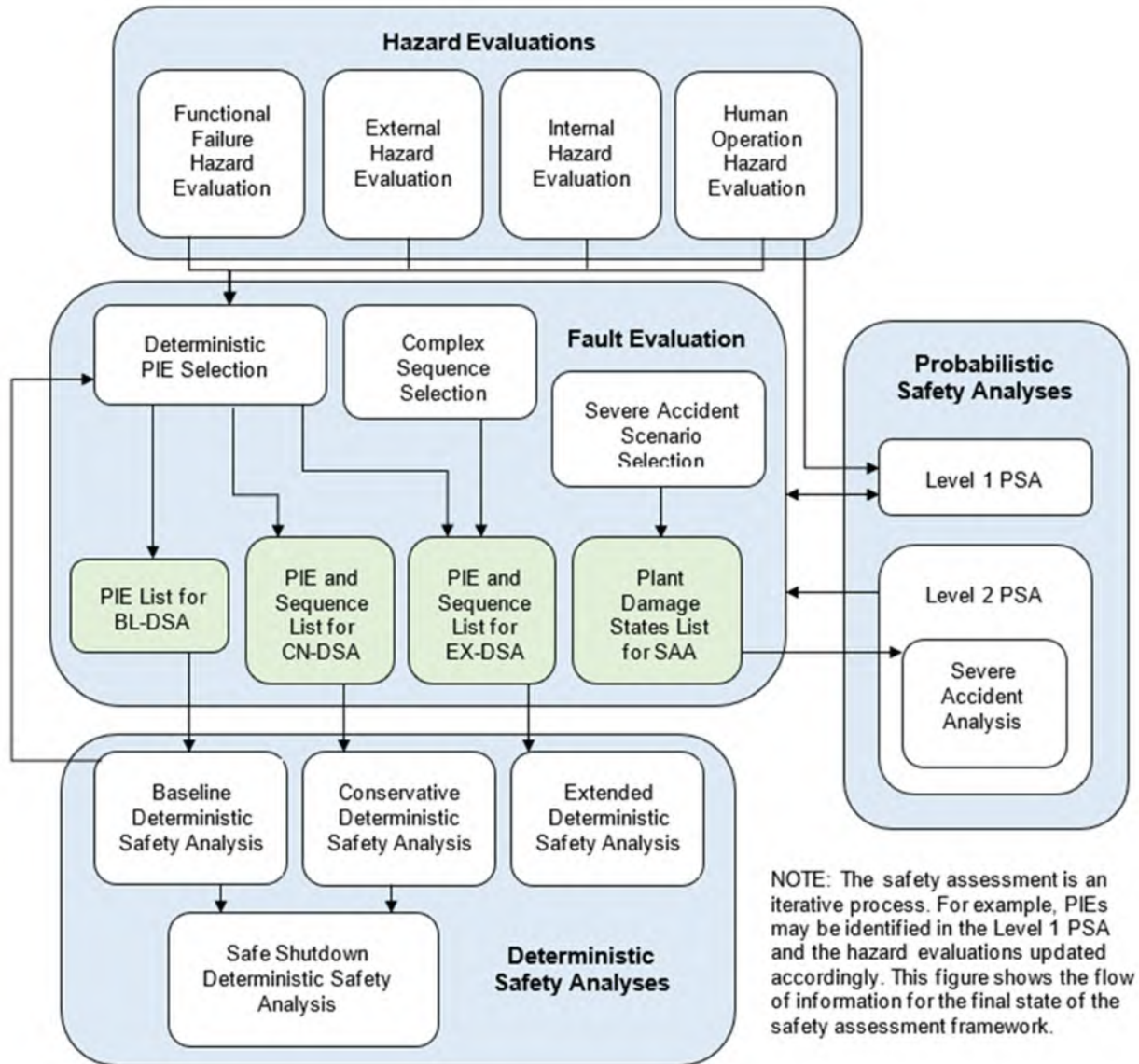


Figure 3.1-2: BWRX-300 Safety Strategy Implementation Process

### **3.2 Classification of Structures, Systems and Components**

The BWRX-300 approach to classifying SSC is consistent with IAEA SSR-2/1 (Reference 3.2-1) and IAEA SSG-30 (Reference 3.2-2) and aligns with CNSC REGDOC-2.5.2 (Reference 3.2-3). Classification of SSC is conducted to identify the importance of the SSC with respect to safety.

This section described how BWRX-300 SSC are classified by:

- Safety Class (SC)
- Seismic Category
- Quality Group

Classification of SSC provides a means for applying appropriate design requirements and establishes a graded approach in the selection of materials, and application of codes and standards used in design, manufacturing, construction, testing and inspection of individual SSC. Sections 3.5 through 3.8 describe the codes and standards applicable to civil, mechanical, I&C, and electrical SSC based on classification.

The classification of SSC also determines the degree of redundancy, diversity, separation, and reliability/availability required as described in Subsection 3.1.7. The requirement for environmental qualification is based on the classification of SSC as described in Section 3.9. In addition, SSC classification informs procurement and quality assurance requirements as discussed in Chapter 17.

#### **3.2.1 Safety Classification Background**

The BWRX-300 approach to classifying SSC by safety class is based primarily on deterministic methods and is directly traceable to the safety functions performed by the SSC. This approach aligns with CNSC REGDOC-2.5.2, Section 7.1, as it reflects:

- Consequences of the SSC failure to perform its safety functions
- Expected frequency of the SSC being called upon to perform its safety functions
- Time following a PIE at which, or the period for which, the SSC may be called upon to perform a safety function

A fundamental element of the BWRX-300 SSC classification approach is the direct correlation between the Defense Line in which an SSC performs a function, and the relative safety importance of that function. Functions are categorized into three safety categories, Safety Category 1, Safety Category 2, and Safety Category 3, with Safety Category 1 being the most important.

##### **3.2.1.1 Primary Function Categorization**

Primary functions are those that directly perform the FSFs in support of DL2, DL3, DL4a or DL4b. Safety Categories are applied to the primary functions as follows:

1. Safety Category 1 is assigned to DL3 primary functions. DL3 functions assure the integrity of the barriers to release, place and maintain the plant in a safe state, and provide independence and diversity for all DL2 and DL4a functions caused by a single failure (and many CCFs). Accordingly, DL3 primary functions are the most important from a safety standpoint.
2. Safety Category 2 is assigned to DL4a primary functions. Both DL2 and DL4a provide a redundant means to address PIEs (generally independent of DL3 functions) and are therefore important from a safety standpoint, although less important than DL3 functions.

DL4a functions are a backup to DL3 functions, in the unlikely event a DL3 functions fails, and therefore have a higher consequence of failure than DL2 functions and are more important from a safety standpoint than DL2 functions.

3. Safety Category 3 is assigned to DL2 and DL4b primary functions as they are relatively the least important. DL4b functions address severe accidents, which are extremely unlikely because failure of both DL3 and DL2 or DL4a functions would have to occur. Accordingly, DL4b functions are considered relatively the least important defense line functions, despite the high consequence of failure.
4. Non-Safety Category is assigned to all other functions.

The assignment of DL4a functions to Safety Category 2, to address the low probability but high consequences of failure, and the assignment of DL4b functions to Safety Category 3, based on the extremely low probability of being called upon, is consistent with CNSC REGDOC-2.5.2, Section 7.1 (Reference 3.2-3), which provides guidance on the treatment of complementary design features called upon to mitigate DEC's.

In addition to categorizing primary functions by the defense line they support, function that provide a supporting role and functions that are not immediately required following a PIE are assigned to a Safety Category as described below and summarized in Table 3.2-1.

### **3.2.1.2 Integral Support Functions**

Integral support functions are functions that support the primary function and are required to be performed concurrently with the primary function (e.g., an HVAC system maintaining the temperature of a space or area within an acceptable range during performance of the primary function (i.e., following the initiating event) to maintain equipment in an acceptable condition).

Integral support functions are considered part of the defense line function (and therefore subject to defense line function "rules," such as independence and diversity) and are assigned the same safety category as the primary function they support.

### **3.2.1.3 Make-Ready Support Functions**

Make-ready support functions are continuously available online functions that maintain the primary function, or a component required to perform the primary function, in a state of readiness but are not required to be performed at the time the primary function is performed. Make-ready functions must have monitoring, such that plant operators would be alerted if the make-ready support function were lost, or the readiness of the primary function or component were compromised. For example, maintaining the temperature of a pool of cooling water within acceptable limits, with monitoring by pool temperature indication is an example of a make-ready support function.

Make-ready functions are not required at the time the primary function is performed and are not considered part of the defense line function (and therefore not subject to defense line function "rules," such as independence and diversity). The primary function would eventually be considered unavailable if the make-ready function were compromised to the extent that the primary function might be compromised. Accordingly, make-ready functions are not required to be assigned the same safety category as the primary function. However, make-ready functions are important and are therefore assigned to safety categories as follows:

- Make-ready functions that support DL3 or DL4a functions are assigned to Safety Category 3
- All other make-ready functions can be assigned to Safety Category N.

#### **3.2.1.4 Delayed Functions**

Delayed functions are primary or support functions that are not required to be performed until sometime after the initiating event. Because there would be ample time during the event to ensure these functions are available, delayed functions are not required to be assigned the same safety category as functions required immediately after the initiating event. If the function is not needed until after 72 hours into the event (but before seven days), it can be classified as Safety Category 2 (instead of Safety Category 1), and if the SSC is not needed until after seven days into the event, it can be classified as Safety Category 3 (instead of Safety Category 1 or Safety Category 2). Delayed functions are not subject to defense line function “rules,” such as independence and diversity.

#### **3.2.1.5 Normal Functions**

Normal functions that perform an FSF during normal plant operation or that maintain key reactor parameters (e.g., reactor pressure and temperature) within normal ranges, and their integral support functions, are assigned to Safety Category 3. Make-ready functions for normal functions can be assigned to Safety Category N. If failure of a normal function would likely result in an initiating event that could challenge an FSF, the function should be assigned to Safety Category 3.

#### **3.2.1.6 Assignment of Safety Class to Components**

Safety Class is assigned to components based on the safety category of the functions they perform as follows:

- Safety Class 1 (SC1) is assigned to SSC that perform a Safety Category 1 function
- Safety Class 2 (SC2) is assigned to SSC that perform a Safety Category 2 function
- Safety Class 3 (SC3) is assigned to SSC that perform a Safety Category 3 function
- Non-Safety Class (SCN) is assigned to all other SSC

Just as with functions, a time-dependency is introduced for components that perform or support DL3 and DL4a functions. Specifically, if the component is not needed until after 72 hours into the event (but before seven days), it can be classified as SC2 (instead of SC1), and if the component is not needed until after seven days into the event, it can be classified as SC3 (instead of SC1 or SC2) because there would be ample time during the event to ensure those components are available. (See Table 3.2-2)

Functions typically have a mission time, which is the time period during which the function is required to be performed. Only SSCs that are required during the mission time of the function are required to be assigned to the safety classes discussed above.

Some component classifications are made for components that perform FSFs but may not be explicitly defined as part of a defense line function. For example:

- Components that are part of design provisions that perform a FSF, whose failure is considered “practically eliminated,” are assigned to SC1. An example is the RPV.
- Components that make up the fission product barriers are assigned to SC1.
- Components that are part of the RCPB are assigned to SC1.

The safety classification of a system is the highest safety classification of any components within the system; however, the component safety classification, and not the system safety classification, defines the design rules applied to components. Assignment of safety

classifications to systems is for convenience in understanding the relative importance of plant systems.

Not all components or parts of a system are necessarily assigned to the same safety class as the system itself. For example, a process system may be classified as SC 1 because one or more of its components support a DL3 function; however, the system may also contain components that support functions associated with other defense lines or components that support no defense line functions. These components are classified in accordance with the defense line functions they support.

Appendix 3A provides a list of the BWRX-300 principal components organized by system and includes their safety classification.

Structures are assigned a safety classification based on the highest safety classification of the components they house or support, excluding components whose failure, due to loss of functionality of the structure, would result in fail-safe performance of the component's safety category function(s). Design rules and performance requirements for structures are derived from their seismic category. Seismic categorization methodology is described in Subsection 3.2.3. The seismic category assigned to a structure is commensurate with its safety classification as listed in Section 3.3, Table 3.3-1.

### **3.2.2 Safety Classification Process**

In alignment with both IAEA and CNSC guidance, this method of classifying the safety significance of SSC is based primarily on deterministic methods because the DL functions are identified using deterministic safety analyses. The deterministic methods are complemented (where appropriate) by probabilistic methods and engineering judgment.

Design rules are then applied to SSC based on their safety classification and the DL functions they support. The safety classification process is iterative with the deterministic and probabilistic safety assessment and is maintained and updated throughout the design phase.

The following outlines the BWRX-300 classification process.

**Review and Definition of PIEs** – Hazard evaluations are performed (as introduced in Section 3.1.6.4.1) to identify hazards with potential to challenge an FSF. The output of these hazard evaluations are potential PIEs.

**Grouping and Identification of Bounding PIEs** – Potential PIEs are grouped by plant effect and occurrence frequency. Bounding or representative PIEs and fault sequences are selected for deterministic safety analyses as described in Chapter 15, Section 15.2.

**Identification of Plant-Specific Safety Functions to Prevent or Mitigate the PIEs** – The deterministic safety analyses are performed and updated iteratively with design activities to establish the plant-specific functions responsible for maintaining the FSFs during PIEs and fault sequences. The identification of plant-specific functions and their assignment to a Defense Line is carried out in the Fault Evaluation described in Chapter 15, Section 15.2 with traceability of each function to each PIE and PIE sequence in which it is credited.

**Safety Categorization of the Safety Functions** – Functions are categorized in accordance with their safety significance and role in performing FSFs. As such, each function receives a safety categorization directly based on its assignment to a DL (as described in Subsection 3.2.1 above).

### **Identification of SSC that Provide the Safety Functions**

Plant-level requirements are created for each DL function and decomposed into system-specific functional requirements to implement the credited DL functions, consistent with the plant

performance modeled in the safety analyses. These requirements are then allocated to the applicable system design description which identifies the components that support the system DL functions.

### **Assignment of SSC to a Safety Class Corresponding to the Safety Category**

Safety Class is assigned to SSC based on the SSC's role in ensuring plant safety, and the defense line and FSF supported as described in Subsection 3.2.1.6 above.

### **Verification of SSC Classification**

The deterministic safety analyses are maintained and updated as the plant design matures. Confirmation of SSC classification is achieved when the deterministic safety analyses models reflect the final plant design and demonstrate compliance to the analysis acceptance criteria (which include rules governing how classified equipment can be credited in each analysis case). This verification is complemented, as appropriate, by insights from the PSA.

### **Identification of Engineering Design Rules for Classified SSC**

Engineering design rules are applied to SSC based on several factors including their SC, their DL role, their status as a pressure boundary component, their role during and following earthquakes, and their operational environment. The design rules establish the scope of codes and standards applied to an SSC, as well as requirements for reliability, diversity, redundancy, and independence applicable to an SSC. These design rules are discussed in Subsection 3.1.7.

### **3.2.3 Seismic Categories**

Seismic Category reflects SSC requirements during and after a seismic event and governs how the SSC is seismically designed and qualified. Seismic Category is assigned based on the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13 (Reference 3.2-3), and CSA N289.1, Clause 5.2.5.2 (Reference 3.2-4) as follows:

1. **Seismic Category A/B** - DL3 functions are credited with remaining operable during and after a seismic event associated with a Design Basis Earthquake (DBE) as defined in Section 3.3.1. Accordingly, SSC that perform or support DL3 functions are categorized as Seismic Category A for passive SSC or Seismic Category B for active SSC. Other SSC that are classified as SC1 per Subsection 3.2.1.6, are categorized as Seismic Category A or B. Any other SSC that are a significant contributor to PSA risk for seismic events are categorized Seismic Category A or B.
2. **Seismic Category RW-IIa** - SSC for management and storage of radiological material that, if released would exceed the dose limits defined in CNSC REGDOC-2.5.2, Section 4.2.1, are categorized as Seismic Category RW-IIa per guidance in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.143, (Reference 3.2-7). These RW-IIa SSC are seismically qualified for one-half of the site-specific DBE. This approach is in accordance with CNSC REGDOC-2.5.2, Section 7.13.1, which permits the use of ASCE/SEI 43 (Reference 3.2-8) graded approach for the seismic classification of SSCs with justification. Based upon the consequences of failure, one-half of the site-specific DBE is justified as it would bound the ground motion spectra for seismic categories identified in ASCE/SEI 43 (Reference 3.2-8) for SSCs used for handling and storage of highly radioactive materials. This justification is described in NEDC-33974P (Reference 3.2-18).
3. **Non-Seismic** - All other SSC are categorized as Non-Seismic and are designed based on applicable non-nuclear requirements, such as those stipulated in the National Building Code of Canada (Reference 3.2-19).

The BWRX-300 Containment and the Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. These two structures are therefore categorized as Seismic Category A structures in the BWRX-300 design per Clause 5.2.5.2 of CSA N289.1 (Reference 3.2-4).

### **3.2.3.1 Seismic Interaction**

SSC that are not Seismic Category A or B but whose failure during a seismic event could adversely affect the ability of any Seismic Category A or B SSC to accomplish its safety function are evaluated for seismic interaction to demonstrate that these SSC:

- Will not collapse or collide with the Seismic Category A and Seismic Category B SSC and will maintain their stability during a DBE or other relevant extreme external hazard event; or
- Impact loads that result from collapse or collision on the Seismic Category A and Seismic Category B SSC are either negligible or smaller than those considered in the design.

In accordance with requirements of Clause 7.2.1.2 of CSA N289.3 (Reference 3.2-6) and Section 6.0 of NEDO-33914 (Reference 3.2-9), interaction evaluations are performed of the Power Block structures and foundations adjacent to the Seismic Category A RB, as described in Subsection 3.3.1.2.8, to ensure:

- These structures and foundations do not collapse to compromise the safety functions of those SSC that are required to remain functional following a DBE or design tornado level event for the first 72 hours.
- The CB structure, which includes the Main Control Room (MCR) does not collapse and result in incapacitating injury to the main control room occupants or prevent their egress to the RB.

Table 3.3-1 in Section 3.3 lists the seismic category and seismic interaction evaluation requirements for structures..

Evaluations for seismic interaction of systems and components is conducted as the design advances and details supporting these evaluations are available.

### **3.2.4 Quality Group**

In alignment with CNSC REGDOC 2.5.2, Section 7.7 (Reference 3.2-3), BWRX-300 pressure-retaining components are designed to ensure they are protected against overpressure conditions, and are classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. The selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. CNSC REGDOC-2.5.2 points to ASME Boiler and Pressure Vessel Code (BPVC) (Reference 3.2-11) to meet the requirements of different classes of pressure-retaining systems, components, piping and their supports.

BWRX-300 design utilizes a Quality Group designation per the guidance in USNRC RG-1.26 (Reference 3.2-10) as a method for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of the component. Items are classified as Quality Group A, B, C or D. The guidance and classification method are used with some clarification based on the unique design of the BWRX-300.

Table 3.2-3 tabulates the design and fabrication requirements for each Quality Group. For mechanical equipment that does not fall within the scope of USNRC RG 1.26 (Reference 3.2-10),

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

appropriate industrial codes and standards are applied. Per CNSC REGDOC-2.5.2, alternative codes and standards may be used with justification and consistent with a graded approach.

Appendix 3A provides a list of the BWRX-300 principal components organized by system and includes their Quality Group. The Quality Group for structures is listed in Section 3.3, Table 3.3-1.

### **3.2.5 References**

- 3.2-1 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design," International Atomic Energy Agency.
- 3.2-2 IAEA Safety Standards Series No. SSG-30, "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," International Atomic Energy Agency.
- 3.2-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.2-4 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.2-5 USNRC Regulatory Guide 1.29, "Seismic Design Classification for Nuclear Power Plants."
- 3.2-6 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.2-7 USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- 3.2-8 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.2-9 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.2-10 USNRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."
- 3.2-11 ASME (BPVC), "Section III," American Society of Mechanical Engineers.
- 3.2-12 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 3.2-13 API 620, "Design and Construction of Large, Welded, Low-Pressure Storage Tanks," American Petroleum Institute.
- 3.2-14 API 650, "Welded Steel Tanks for Oil Storage," American Petroleum Institute.
- 3.2-15 AWWA D100-11, "Welded Carbon Steel Tanks for Water Storage," American Water Works Association.
- 3.2-16 ASME B96.1, "Welded Aluminum-Alloy Storage Tanks," American Society of Mechanical Engineers.
- 3.2-17 TEMA, "Standards of the Tubular Exchanger Manufacturers Association," Tubular Exchange Manufacturers Association.
- 3.2-18 NEDC-33974P, "BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report," GE-Hitachi Nuclear Energy Americas, LLC.



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.2-19 Canadian Commission on Building and Fire Codes, "National Building Code of Canada,"  
National Resource Council of Canada.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.2-1: Safety Category for Functions Based on Defense Line Assignment**

Safety Category	Defense Line 3 Functions	Defense Line 4a Functions	Defense Line 2/4b Functions	Normal Functions
1	<ul style="list-style-type: none"> <li>Primary and Integral support functions required within the first 72 hours of an event</li> </ul>			
2	<ul style="list-style-type: none"> <li>Primary and integral support functions required after 72 hours but before 7 days after an event</li> </ul>	<ul style="list-style-type: none"> <li>Primary and integral support functions required within the first 7 days of an event</li> </ul>		
3	<ul style="list-style-type: none"> <li>Primary and integral support functions required after 7 days after an event</li> <li>Make-ready support functions</li> </ul>	<ul style="list-style-type: none"> <li>Primary and integral support functions required after 7 days</li> <li>Make-ready support functions</li> </ul>	<ul style="list-style-type: none"> <li>All primary and integral support functions</li> </ul>	<ul style="list-style-type: none"> <li>Normal functions that perform a fundamental safety function</li> <li>Normal functions that maintain key reactor parameters (e.g., pressure and temperature) within normal ranges</li> <li>Integral support functions</li> </ul>
N			<ul style="list-style-type: none"> <li>Make-ready support functions</li> </ul>	<ul style="list-style-type: none"> <li>Make-ready support functions</li> </ul>

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.2-2: Safety Class for SSC**

Safety Class	Safety Category 1 Functions	Safety Category 2 Functions	Safety Category 3 Functions	Safety Category N Functions	Other
1	<ul style="list-style-type: none"> <li>SSCs required within first 72 hours of event</li> </ul>				<ul style="list-style-type: none"> <li>Components that are part of design provisions that perform a FSF, whose failure is considered "practically eliminated"</li> <li>Components that make up the fission product barriers</li> <li>Components that are part of the reactor coolant pressure boundary</li> </ul>
2	<ul style="list-style-type: none"> <li>SSCs required after 72 hours but before 7 days</li> </ul>	<ul style="list-style-type: none"> <li>SSCs required within first 7 days of event</li> </ul>			
3	<ul style="list-style-type: none"> <li>SSCs required after 7 days</li> </ul>	<ul style="list-style-type: none"> <li>SSCs required after 7 days</li> </ul>	<ul style="list-style-type: none"> <li>All SSCs</li> </ul>		
N				<ul style="list-style-type: none"> <li>All SSCs</li> </ul>	

Note: Only SSCs that are required during the mission time of the function are required to be assigned to the safety classes discussed above.

**Table 3.2-3: Codes and Standards for Pressure-Retaining Equipment**

Quality Group	ASME BPVC Section III Code Classes	Pressure Vessels and Heat Exchangers <sup>(4)</sup>	Pipes, Valves, and Pumps	Storage Tanks 0-103 kPaG (0-15 psig)	Storage Tanks Atmospheric	ASME BPVC Section III Component Supports	Non-ASME BPVC Section III Component Supports	Core Support Structures and Reactor Internals	Containment Boundary
A	1	NCA and NB	NCA and NB	—	—	NCA and NF	—	—	—
B	2	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NF	—	—	—
	MC	—	—	—	—	—	—	—	NCA and NE <sup>(1)</sup>
	CS	—	—	—	—	—	—	NCA and NG	—
C	3	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NF	—	—	—
D	—	ASME BPVC Sect. VIII Division 1	ASME B31.1 for piping and valves <sup>(2)</sup>	API 620 or equivalent <sup>3</sup>	API 650 AWWA D100-11 ASME B96.1 or equivalent <sup>(3)</sup>	—	Manufacturer Specified Standards, e.g., ASME B31.1, AISC	—	—

(1) Excluding the Steel-plate Composite Containment Vessel. See Section 3.5.3 for applicable codes and standards.

(2) For pumps classified in Quality Group D, the ASME BPVC, Section VIII, Division 1 is used as a guide in determining the wall thickness for pressure-retaining parts and in sizing the cover bolting.

(3) Tanks are designed to meet the intent of American Petroleum Institute (API) Standard 620 (Reference 3.2-13), API 650 (Reference 3.2-14), American Water Works Association (AWWA) (Reference 3.2-15), and/or ASME B96.1 standards (Reference 3.2-16, as applicable).

(4) For Tubular Exchanger Manufacturers Association (TEMA)-style heat exchangers, both the ASME Code and TEMA standard (Reference 3.2-17) are considered. Other heat exchanger design styles/configurations are not subject to the TEMA standard.

(5) Acronyms used in Table 3.2-2 refer to the ASME BPVC Section III (Reference 3.2-11) subsections as follows:

- Subsection NCA-General Requirements for Division 1 and Division 2
- Division 1 Subsections:
  - Subsection NB – Class 1 Components

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Subsection NCD - Class 2 and 3 Components
- Subsection NE- Metal Containment (MC)
- Subsection NF – Supports
- Subsection NG – Core Support Structure (CS)

### **3.3 Protection Against External Hazards**

Complying with Section 7.4.2 of CNSC REGDOC-2.5.2 (Reference 3.3-1), the BWRX-300 design considers natural and human-induced external hazards that may be linked with significant radiological risk. This section discusses external hazards relevant to the DNNP site and the BWRX-300 approach to prevent and mitigate their effects on Safety Class 1 (SC1) Structures, Systems and Components (SSC). SC2/SC3 SSC that are credited in the fault evaluation with mitigating fault sequences initiated by external hazards, and SSC whose failure can affect the structural integrity or safety class functions of adjacent SC1 SSC are also protected against external hazards.

The determination of the external hazards considered in the BWRX-300 design relies on the collection of the geotechnical, seismological, hydrological, hydrogeological, and meteorological reference data, and human-induced external events presented in Chapter 2, Section 2.2, Section 2.4, Section 2.5, Section 2.6 and Section 2.7. For external hazards, the main protection is provided by the civil structures. The design against external hazards is such that a design basis external hazard does not lead to a Design Basis Accident (DBA) or a Beyond Design Basis Accident (BDBA). Significant safety margins are included in the evaluation of the design basis external hazards and the associated design aspects to ensure a conservative design. Assurance that the overall reactor plant is resilient to external hazards is provided by the demonstration that SSC do not fail when subject to these hazards and generated loadings. Demonstration of the adequacy of protection measures is provided in the applicable PSAR chapters covering the design of SSC.

Malevolent acts considered in the robustness design are discussed in Subsection 3.3.7.4.

Protection and mitigation methods considered in the design are in line with the design safety objectives and Defence-in-Depth (D-in-D) concept discussed in Subsections 3.1.1 and 3.1.6, respectively. They include the use of physical separation, barriers/shielding, qualification of equipment and instrumentation for the hazards environment and monitoring programs to preclude unacceptable radiation releases following accidents due to external hazards.

When applicable, loads generated by external hazards are considered in the BWRX-300 design following requirements in Section 7.15.1 of CNSC REGDOC-2.5.2 and CSA N291 (Reference 3.3-2). Combination of loads from randomly occurring individual external hazards is considered in the design to ensure structures are adequately protected against external hazards.

A principal safety objective of the BWRX-300 Safety Strategy is the demonstration that the overall reactor plant design is resilient to hazards through D-in-D. This means that the design provisions optimize protection to provide the highest level of safety that can reasonably be achieved such that relevant dose targets on-site and off-site are met and the resilience of the reactor plant to external hazards reduces risk. The process of demonstrating that the reactor plant is resilient starts with the systematic identification of Postulated Initiating Event (PIEs) with a potential to challenge a fundamental safety function, and to organize them into the fault list developed as per Chapter 15, Section 15.2. Combinations of randomly occurring individual events are considered in these evaluations in accordance with requirements in CNSC REGDOC-2.5.2, Section 7.4.3. Deterministic and probabilistic safety analyses are then performed as discussed in Chapter 15, Sections 15.5 and 15.6 to confirm the design adequacy and its resilience to these hazards. Summary of results of the safety assessments are presented in Section 15.7.

#### **3.3.1 Seismic Design**

For seismic design, BWRX-300 SSC are categorized as Seismic Category A, Seismic Category B, Seismic Category RW-IIa and/or Non-Seismic Category as discussed in Subsection 3.2.3. This

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

seismic categorization reflects SSC's functional and performance requirements during or after a seismic event and impacts their design.

Following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, CSA N289.1 (Reference 3.3-3) and U.S. NRC RG 1.208 (Reference 3.3-4), Seismic Category A and Seismic Category B SSC are seismically qualified to withstand the effects of a Design Basis Earthquake (DBE) that is developed:

1. Based on the geological, seismological, and geotechnical conditions at the site described in Chapter 2, Section 2.7
2. Following the performance-based approach of ASCE/SEI 43 (Reference 3.3-5) Section 2 for development of DBE for seismic design of structures achieving a target performance goal of  $1E-5$  per year
3. Meets the minimum earthquake requirements of CSA N289.3 (Reference 3.3-6), Clause 4.2

The development of the 5% damped Acceleration Response Spectra (ARS) defining the amplitude and frequency content of the bounding site-specific DBE input ground motion used for the seismic qualification of Seismic Category A and B SSC is discussed in Subsection 3.3.1.1.

Table 3.3-1 provides the seismic categorization of BWRX-300 structures. Per Subsection 3.2.3, the containment, and Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. As a result, the integrated RB structure, which consists of the RB, containment and containment internal structures is the only structure categorized as Seismic Category A in the BWRX-300 design. As shown in Table 3.3-1, the seismic design of the Seismic Category A structures considers Limit State LS-D response defined in Table 1-2 of ASCE/SEI 43 as essentially elastic response without any significant permanent deformation. According to U.S. NRC RG 1.208, this ensures a consistent level of safety from earthquake-caused failures defined by level of response resulting in an onset of significant inelastic deformations with a probability of unacceptable performance:

- Less than 1% for a DBE ground motion level
- Less than 10% for ground motion with 1.5 times the DBE intensity

The Radwaste Building (RWB) which processes and houses liquid, solid and gaseous radwaste is categorized as Seismic Category RW-IIa as shown in Table 3.3-1. The remaining BWRX-300 Power Block structures, which consist of the Control Building (CB), Turbine Building (TB) and Reactor Auxiliary Bay (See Chapter 1, Appendix A, Figure A1.4-1) are categorized as Non-Seismic.

Due to their proximity to the Seismic Category A RB, the RWB, CB, TB and Reactor Auxiliary Bay are evaluated for interaction with the integrated RB structure per the requirements in SSR-2/1 (Reference 3.3-7), Section 5.19, as discussed in Subsection 3.2.3.1. The interaction evaluation methodology is presented in Subsection 3.3.1.2.8. Table 3.3-1 summarizes the seismic design basis for the BWRX-300 structures based on their seismic categories. Per Table 3.3-1, the RW-IIa structures are designed per CSA N291 and U.S. NRC RG 1.143 (Reference 3.3-8), while Non-Seismic Category structures are designed in accordance with the National Building Code of Canada (NBC) (Reference 3.3-9). The primary focus of this section is on the seismic qualification of Seismic Category A and Seismic Category B SSC. The seismic design of the RW-IIa and Non-Seismic Category structures is further discussed in Chapter 9B, Section 9B.3.

Seismic robustness of Seismic Category A structures is evaluated for a Design Extension Condition (DEC) Checking Level Earthquake (CLE) as described in Subsection 3.5.6.1.

The BWRX-300 design considers Operating Basis Earthquake (OBE) and Site Operating Earthquake loads as 1/3 of the DNNP site-specific DBE. Per Appendix S to 10 CFR 50 (Reference 3.3-10), design load combinations that consider OBE and Site Operating Earthquake loads are not required, except for the design of metal containment components where the OBE loads are considered for post-flooding condition and cyclic loading considerations, as noted in Table 9B-1 in Chapter 9B. OBE is not used as reference earthquake for the BWRX-300 DNNP plant shutdown.

The DNNP BWRX-300 seismic instrumentation is discussed in Subsection 3.3.1.5. As described in Subsection 3.3.1.5, the criteria for seismic instrumentation, plant shutdown, evaluation and inspection are in accordance with the guidelines of CSA N289.5 (Reference 3.3-11) and Clause 6.5 of CSA N289.1.

#### **3.3.1.1 Bounding Seismic Design Parameters**

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the design of the BWRX-300 is based on DNNP site-specific geotechnical and seismic inputs. Bounding seismic design parameters are developed based on the data that was available prior to the completion of the characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative site-specific seismic inputs adequately address uncertainties related to the use of incomplete (preliminary) characterizations of the DNNP geotechnical and seismic conditions.

The 5% damped spectra defining the magnitude and frequency content of the DNNP bounding site-specific design ground motion are developed based on the results of probabilistic Site Response Analysis (SRA) presented in Subsection 3.3.1.1.2 using as input the dynamic subgrade properties dynamic subgrade properties described in Subsection 3.3.1.1.1.

The results of the probabilistic SRA are also used for the development of bounding stiffness and damping properties of subgrade materials that are compatible with the free-field strains generated by a typical design level earthquake event.

The bounding DBE ground motion response spectra in Subsection 3.3.1.1.3 and the bounding strain-compatible dynamic subgrade profiles discussed in Subsection 3.3.1.1.6 provide a conservative seismic design that adequately address the aleatory variabilities and epistemic uncertainties in the geotechnical properties of the DNNP site.

Five sets of ground motion time histories compatible to the bounding DBE ground motion response spectra are developed, as described in Subsection 3.3.1.1.4, for use as input for the linear seismic Soil-Structure Interaction (SSI) analysis.

##### **3.3.1.1.1 Bounding Dynamic Subgrade Properties**

The bounding seismic design parameters are developed using dynamic properties for the subgrade rock, in-situ soil, and engineered fill that are determined based on the data obtained from multiple geotechnical investigations that were completed at the vicinity of the DNNP site prior to the geotechnical site investigations and laboratory tests described in Chapter 2 Section 2.7.3.

For use as input for the probabilistic SRA described in Subsection 3.3.1.1.2, bounding subgrade dynamic profiles are developed reflecting anticipated as-built conditions at the site after construction of the BWRX-300 SMR that include compacted fill from about elevation 80 to 82 m Canadian Geodetic Datum (CGD) to the final grade at elevation 88 m CGD. The layering of the in-situ soil materials is determined based on the stratigraphy obtained from the studies presented in:

- NK054-REP-01210-00098 (Reference 3.3-12) providing data from multiple borings near the proposed BWRX-300 SMR site (B-104, B-113, B-116, and B-118), and



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- NK054-REF-01210-0418696 (Reference 3.3-13) providing data from deeper borings close to the BWRX-300 SMR (AMC-03ALT).

It is anticipated that the loose surficial soil materials that are not competent for supporting the heavy foundations of the power block buildings and have potential for liquefaction during earthquakes will be excavated and replaced with an engineered fill obtained by reconditioning and compacting the in-situ soils from the fill layer, surficial lacustrine layer, and upper till materials excavated from the upper 6 to 8 m of the site. Results of compaction tests of the in-situ soil materials in 2009 NK054-REP-07730-00005 (Reference 3.3-14) are used as basis for development of the engineered fill dynamic properties.

The probabilistic SRA, described in Subsection 3.3.1.1.2, explicitly consider the epistemic uncertainties in the estimation of subgrade dynamic properties by using 50th percentile Best Estimate (BE), 10th percentile Lower Realization (LR), and 90th percentile Upper Realization (UR) values for the shear wave velocities and kappa representing the dissipation of the energy for the site. For the different subgrade materials, standard deviation for the natural log of the shear wave velocity is assigned to adequately define the aleatory variability of subgrade dynamic stiffness properties.

The profile of bounding rock dynamic properties is developed directly from the recommended shear wave velocity profiles in 2012 NK054-REF-01210-0418696 (Reference 3.3-13). The Base Case values and variations for dynamic properties of rock are presented in Table 3.3-2. The compression wave velocities, shear wave velocities, and Poisson's ratio for the bedrock rock units are obtained from the measured values from 2012 NK054-REF-01210-0418696 (Reference 3.3-13) without modification. The rock Poisson's ratio was calculated from the measured compression and shear wave velocities following the recommendation of the NEDO-33914 (Reference 3.3-15).

The profile of base case dynamic properties presented in Table 3.3-2 considers the following:

1. The "Top of Bedrock Rock" elevation is 64.1 m CGD with a  $\sigma_{TOR}$  of  $\pm 1$  m
2. The variation in the rock layers assumes  $\pm 2$  m
3. The  $\sigma_{\mu \ln}$  represents the epistemic uncertainty for estimating LR (10th percentile) and UR (90th percentile) profiles
4. The  $\sigma_{\ln V_s}$  represents the aleatory uncertainty for randomization of the shear wave velocities.

Epistemic uncertainty in the distribution of the shear wave velocity profiles ( $\sigma_{\mu \ln}$ ) was estimated based on the range of  $V_s$  values measured in each bedrock layer; however, the estimated values were lower than the typical estimate of 0.35 in the 2013 EPRI TR-1025287 (Reference 3.3-16). Based on a comparison with the estimated  $\sigma_{\mu \ln}$  values, a  $\sigma_{\mu \ln}$  of 0.10 is selected based on the similar results from all three borings, as described in the 2012 NK054-REF-01210-0418696 (Reference 3.3-13). Using a higher  $\sigma_{\mu \ln}$  value was not justified by the site data. Aleatory uncertainty considers a standard deviation for the natural log of the shear wave velocity ( $\sigma_{\ln V_s}$ ) of 0.15 for the bedrock layers based on the 2013 EPRI TR-1025287 (Reference 3.3-16).

Table 3.3-3 presents the small-strain dynamic properties of the engineered fill and the in-situ soil. The small-strain values of the soil materials are estimated from the measured SPT N60 values provided in the NK054-REF-01210-0418696 (Reference 3.3-13) and the NK054-REP-01210-00098 (Reference 3.3-12). Three sets of shear wave velocities are estimated for each soil layer using the average, lowest, and highest N60 values. The results for the average, lower, and upper estimates were then combined using weights of 0.4, 0.3, and 0.3, respectively, to approximate a normal distribution, per the 2013 EPRI TR-1025287 (Reference 3.3-16).

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The uncertainty in the estimates of soil  $V_s$  is considered using a  $\sigma_{\mu \ln}$  of 0.35 to 0.40. Per recommendations in the 2013 EPRI TR-1025287 (Reference 3.3-16), a value of 0.35 is intended for sites with limited shear wave velocity data while a value of 0.50 is appropriate for a site without shear wave velocity data. The selected  $\sigma_{\mu \ln}$  values generally cover the range of estimated  $V_s$  values in each soil layer at the 10th and 90th percentile.

Dynamic fill properties are estimated from the  $N_{60}$  values. Two correlations are used to estimate  $V_s$  for the  $N_{60}$  values, per the 2012 PEER Report 2012/08 (Reference 3.3-17). The selected  $V_s$  correlations use the  $N_{60}$  values and are appropriate for fill using a range of soils. The average of the two correlations was used as the shear wave velocity in each fill layer. A  $\sigma_{\mu \ln}$  of 0.40 was selected. The selected  $\sigma_{\mu \ln}$  value is considered reasonable due to the limited information on the fill materials. A  $\sigma_{\ln V_s}$  value of 0.25 is used for the fill and upper till and a value of 0.15 is used for the deeper in-situ soil layers.

The BE, LR, and UR variations of the kappa parameter, used to establish consistent damping ratios for the rock layers at the site are presented in Table 3.3-4. The kappa value was estimated following the guidance of the 2013 EPRI TR-1025287 (Reference 3.3-16) for CEUS firm rock profiles with a thickness of less than 1000 m and a total standard deviation of 0.47 for kappa based on the 2014 PEER Report No. 2014/12 (Reference 3.3-18).

The BE, LR and UR of the shear wave velocity profile representing the assumed as-built conditions are presented in Figure 3.3-1.

The dynamic subgrade stiffness properties of in-situ soil and engineered fill materials in Table 3.3-3 correspond to small-strain levels. To account for the nonlinearity of the engineered fill and in-situ soil materials. The following two sets of strain-dependent property curves are recommended in EPRI TR-1025287 (Reference 3.3-16, Section B-3.3):

- EPRI curves from the 1993 EPRI TR-102293, "Guidelines for determining design basis ground motions (Reference 3.3-19)
- Peninsular Range curves, Silva, W.J., N. Abrahamson, G. Toro and C. Costantino. (1996). Description and validation of the stochastic ground motion model (Reference 3.3-20)

The Peninsular Range curves are used for the development of bounding seismic design parameters to account for the strain-dependance of the soil and engineered fill dynamic stiffness and damping properties. The EPRI curves are not considered because the results of SRA indicated excessive softening of the soil and fill layers which can result in unconservative estimates of the seismic response at the ground surface, per the 2013 EPRI TR-1025287 (Reference 3.3-16, Section 5.0, and Figure 5-7).

#### **3.3.1.1.2 Site Response Analyses**

Probabilistic Site Response Analyses (SRA) are performed to accommodate the effects of overlying materials on the seismic hazard considering the epistemic uncertainties and aleatory variabilities in the site parameters to preserve the desired hazard levels and performance goals per requirements of CSA N289.2 (Reference 3.3-21) and regulatory guidelines of U.S. NRC RG 1.208. These SRA consider as-built conditions at the DNNP site after the excavation, construction, and backfilling. The equivalent linear approach is used for the SRA to account for the non-linear response of the soil. Curves representing the shear modulus reduction ( $G/G_{max}$ ) and damping of the soil materials as a function of strain are used to iteratively adjust the shear modulus and damping ratio of the soil based on the calculated effective soil shear strain until convergence is obtained.

As discussed in Subsection 3.3.1.1.1, epistemic uncertainties in the shear wave velocities and the dissipation of energy for the site represented by the coefficient kappa are explicitly considered in

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

the evaluation of DNNP bounding seismic parameters. To account for the epistemic uncertainties, the probabilistic SRA consider three sets of values BE, LR, and UR for shear wave velocity, presented in Figure 3.3-1 and kappa values presented in Table 3.3-4, resulting in a total of 9 sets of base case analyses. Per 2013 EPRI TR-1025287 (Reference 3.3-16), weight factors of 0.4, 0.3, and 0.3 are assigned for the BE, LR and UR cases, respectively. The cases considered for the epistemic uncertainties and their associated weight factors are presented in Figure 3.3-2.

The SRA consider aleatory variabilities related to variations in layer thicknesses including rock depth, shear wave velocities, non-linear degradation curves for the engineered backfill and soil layers, and rock damping. The aleatory variabilities are included in the site response analysis by randomization of the BE, LR and UR shear wave velocity base case profiles, using a sample size of 60 with log-normal distributions.

The range of simulated shear wave velocities is limited to two log-standard deviations above and below the specified median value to bound the randomized profiles within physically plausible limits.

Toro's site variation model (Reference 3.3-22) is used for the randomization of the thickness of soil and rock layers. The site variation model parameters are modified to capture a value of 1 m for the variation of rock depth without regards to the thickness variation in the soil layers above or the rock layers below the rock top elevation. This is a reasonable approximation since:

- The top layer is engineered backfill
- The effects of the thickness variations within the soil and rock layers on the site response are insignificant compared to the variation of the elevation of the rock and soil interface

Figure 3.3-3 shows the suite of 60 random shear wave profiles that include the thickness variations obtained from the randomization of the BE shear wave and BE kappa value (BE-BE) base case profile. The thick black line in the plot designates the resulting mean profile.

The curves representing the shear modulus reduction ( $G/G_{max}$ ) and damping of the soil materials with strain are randomized into 60 realizations with correlated log-normal distribution using the Darendeli model (Reference 3.3-23). The damping of subgrade materials is limited to 15% in accordance with the regulatory guidance of ASCE/SEI 4 (Reference 3.3-24), Section C5.2 and U.S. NRC RG 1.208, Appendix E. Figure 3.3-4 shows examples of randomized modulus reduction and material damping curves. The thick black lines in these plots designates the resulting mean curves.

Approach 1, from the approaches defined in NUREG/CR-6728 (Reference 3.3-25), is implemented for the SRA, where the reference site Uniform Hazard Response Spectra (UHRS) with Mean Annual Probability of Exceedance (MAPE) of  $1E-3$ ,  $1E-4$  and  $1E-5$ , are directly used as input control motions and propagated from the bedrock with reference shear wave velocity of 2,800 m/sec through the randomized subgrade profiles. This allows the 5% damped ARS results of Approach 1 SRA to be directly used for the development of the UHRS representing the seismic hazard at the horizons of interest.

Approach 1 is selected as appropriate approximation for the purposes of development of bounding seismic parameters using a preliminary site information.

The reference site UHRS at  $1E-03$ ,  $1E-04$ , and  $1E-05$  MAPE levels are developed using the results of the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26). Between the different options considered in this PSHA, Option 2 for CAV filtering of magnitudes 5 and above is used as input for the Approach 1 SRA, as it provides the greater seismic hazard. Figure 3.3-5 shows the bedrock UHRS used as input for the SRA.

Using the random vibration theory, power spectral density functions for the reference site motions are calculated iteratively from the input UHRS and propagated throughout the randomized shear wave profiles to calculate power spectral density functions at the horizons of interest. 5% damped ARS at each horizon of interest are then calculated from their corresponding power spectral density functions implementing the random vibration theory approach.

For each of the 9 base cases shown in Figure 3.3-2 and MAPE considered, log-mean ( $\mu_i$ ) and log-Standard Deviation ( $\sigma_i$ ) 5% damped ARS results are calculated from the SRA of the 60 random profiles. UHRS representing the mean estimate of the seismic hazard at the horizons of interest are calculated by applying weight factors ( $w_i$ ) to the log mean ARS results from the different base case analyses as follows:

$$UHRS = \sum_i w_i \mu_i$$

Figure 3.3-6 and Figure 3.3-7 show with thick solid red lines the MAPE 1E-4 and 1E-5 UHRS representing the seismic hazard at the ground and top of rock surfaces, respectively, together with the corresponding log-mean ARS calculated from the analyses of 9 base cases.

Log-Standard Deviation values  $\sigma_T$  and  $\sigma_{Ep}$  are calculated as follows, representing the composite (total) uncertainty and epistemic uncertainty of the calculated hazard at the horizons of interest, respectively:

$$\sigma_T = \sqrt{\sum_i w_i ((\mu_i - \mu_T)^2 + \sigma_i^2)}$$

$$\sigma_{Ep} = \sqrt{\sum_i w_i (\mu_i - \mu_T)^2}$$

Figure 3.3-8 and Figure 3.3-9 present the composite and epistemic uncertainties for the MAPE 1E-4 and 1E-5 seismic hazard for the responses at the ground and top of rock surfaces, respectively. The figures also show the log-Standard Deviation of the ARS results for the 9 base cases.

Upper Bound (UB) estimates of the UHRS ( $UHRS_{UB}$ ) are developed to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology by applying one epistemic log-normal Standard Deviation ( $\sigma_{Ep}$ ) increments to the mean hazard estimate UHRS as follows:

$$UHRS_{UB} = UHRS \times e^{\sigma_{Ep}}$$

Figure 3.3-6 and Figure 3.3-7 show with thick dashed lines the UB UHRS for MAPE 1E-4 and 1E-5 representing the UB estimates of the seismic hazard at the ground and top of rock surfaces, respectively.

### 3.3.1.1.3 Design Basis Seismic Ground Motion Response Spectra

Acceleration response spectra at 5% damping define the amplitude and frequency content of the BWRX-300 design ground motion consistent with Clause 4.3 of CSA N289.3. In accordance with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the horizontal ground motion design spectra are developed following the methodology specified in Section 2 of ASCE/SEI 43 using the UHRS results with annual probability of exceedance of 1E-4 and 1E-5 per year.

Additional requirements for developing the site-specific DBE for the design of the deeply embedded Seismic Category A integrated RB structure are provided in Section 5.2.2 of NEDO-33914 Revision 2 (Reference 3.3-15).

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The following horizontal and vertical spectra define the amplitude and frequency content of the DNNP site-specific DBE ground motion for the SSI analysis of the BWRX-300 deeply embedded RB structure:

1. Foundation Input Response Spectra (FIRS) defining the DBE ground motion at bottom of RB Foundation.
2. Performance Based Surface Response Spectra (PBSRS) defining the DBE ground motion at the finished plant grade elevation.
3. Performance Based Intermediate Response Spectra (PBIRS) defining the DBE ground motion at intermediate embedment depth elevation established, following the guidelines in NEDO-33914 Revision 2, Section 5.2.2 at the top of the rock elevation having a significant contrast between rock and overlaying soil shear wave velocities.

The purpose of PBIRS is to ensure the ground motions used as input for the SSI analyses of deeply embedded structures are adequate throughout the depth of the embedment.

Horizontal FIRS, PBSRS and PBIRS are developed following the performance-based approach criteria of ASCE/SEI 43, Section 2 for DBE with a target performance goal of  $1E-5$ . Instead of using UHRS representing the mean estimate of the seismic hazard as mandated by ASCE/SEI 43, the bounding FIRS, PBSRS and PBIRS are conservatively developed using the  $1E-4$  and  $1E-5$  MAPE UHRS representing UB estimates of the seismic hazard. These UB UHRS are developed as described in Subsection 3.3.1.1.2 to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology. The resulting horizontal Ground Motion Response Spectra (GMRS) are further adjusted to meet the minimum required response spectra requirement using the generic spectrum in CSA N289.3, Clause 4.3.2 anchored at the minimum peak ground acceleration value of  $0.1g$ .

Horizontal reference site hard rock GMRS is also developed following the ASCE/SEI 43 performance-based approach using the UHRS obtained from the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26) representing the reference site hazard with MAPE of  $1E-4$  and  $1E-5$ . This reference site hard rock spectrum is used to conservatively neglect the de-amplifications of the reference hazard motion as it propagates through the rock column. A single rock design ground motion response spectrum is developed as a conservative representation of the amplitude and frequency content of the horizontal rock GMRS by enveloping, as shown in Figure 3.3-10 the three GMRS representing the seismic hazard at FIRS, PBIRS and reference site hard rock horizons.

The horizontal PBSRS representing the amplitude and frequency content of the design motion at the ground surface are increased to conservatively account for the uncertainties in the soil column properties that may result in spectral peak shifts by connecting the spectral peaks in the PBSRS at frequencies of 8.3 Hz and 20.4 Hz using linear interpolation in the logarithmic space.

Figure 3.3-11 presents the development of the enveloping 5% damped PBSRS representing the amplitude and frequency content of the horizontal design ground motion at the finished grade elevation.

Vertical rock GMRS and PBIRS are developed by applying frequency-dependent Vertical-over-Horizontal (V/H) ratios to the bounding horizontal spectra, in accordance with the requirements of CSA N289.3, Clause 4.3.3.3 and U.S. NRC RG 1.208.

The rock V/H ratios that are used for calculation of vertical rock GMRS, are constructed using the CEUS hard rock V/H ratios from NUREG/CR-6728 (Reference 3.3-25). The vertical PBSRS are calculated using soil V/H that are constructed following the methodology for CEUS soil sites using the procedure outlined in Appendix J of NUREG/CR-6728 (Reference 3.3-25). The rock and soil

V/H ratios used for calculation of the bounding vertical ground motion design spectra are presented in Figure 3.3-12.

Figure 3.3-13 presents the site-specific horizontal and vertical rock Design Ground Response Spectrum (DGRS) and PBSRS defining the bounding design ground motion for the seismic analysis of the BWRX-300 Seismic Category A structures and for the seismic interaction evaluations discussed in Subsection 3.3.1.2, and compares these bounding values to the corresponding ground motion response spectra developed using the latest available geotechnical and seismological data (described in Chapter 2, Section 2.7), which were not available at the time of development of the bounding seismic design parameters.

The bounding horizontal and vertical peak ground accelerations for the rock design ground motion is 0.31 g. For the surface ground motion, the bounding peak accelerations are 0.532 g and 0.516 g for the horizontal and vertical directions, respectively. Peak ground acceleration values are defined as the ground motion acceleration values at 100 Hz.

NEI checks are performed following the procedure described in Section 5.3.4 of NEDO-33914 Revision 2 to ensure the ground motion used as input for the deterministic SSI analyses of deeply embedded RB structure at the RB foundation bottom elevation meets the regulatory guidance of U.S. NRC DC/COL-ISG-017 (Reference 3.3-27) to be hazard consistent with the results of probabilistic SRA. Horizontal and vertical rock design GMRS input motions are propagated upward through the strain-compatible soil profiles, developed as described in Subsection 3.3.1.1.6, from the bottom of foundation to the profiles surface. The envelope of the 5% damped ARS results for the responses at surface of the profiles are compared to the PBSRS. When the enveloped ARS do not meet or exceed the PBSRS, the design spectra are augmented to ensure that the augmented motion satisfies the NEI check. The augmented spectra are further increased to smooth spectral peaks and fill the valleys. Figure 3.3-14 presents the NEI check augmented and smoothed horizontal and vertical 5% damped spectra defining the amplitude and frequency content of the SSI input control motion applied to the SSI model at the RB foundation bottom.

As shown in Figure 3.3-13, in the frequency range of 0.5 to 50 Hz, which is of interest for the seismic design, the bounding horizontal Rock DGRS and PBSRS envelop the corresponding updated design response spectra discussed in Chapter 2, Section 2.7. Exceedances can be observed in the vertical Rock DGRS of up to 10% for frequencies up to 15 Hz. There are also exceedances in the vertical PBSRS of up to 20% for frequencies ranging from 2 Hz to 30 Hz.

The results of the sensitivity evaluation discussed in Chapter 9B Appendix 9B.C indicate the conservatism introduced in the bounding DNNP site-specific seismic design by using the enhanced input ground motion in Figure 3.3-14. Considering this and the other sources of conservatism in the analysis inputs and methodology as well as the considerable margins in the site-specific design of the RB integrated structures demonstrated by the structural design evaluations discussed in Chapter 9B Appendices 9B.E – 9B.G, the conclusions of the bounding seismic SSI evaluations are not expected to be affected by the relatively small exceedances of bounding ground motion Design Response Spectra observed in Figure 3.3-13.

#### **3.3.1.1.4 Design Time Histories**

Design ground motion acceleration time histories used as input to the seismic SSI analyses of RB are developed by spectral matching seed ground motion records to the ground motion design response spectra presented in Figure 3.3-14. Per the guidelines of NEDO-33914 Revision 2, Section 5.2.3, five sets of three design motion time histories, in the two horizontal and in the vertical directions, are developed for the design to mitigate uncertainties due to the phasing of the time history frequency components.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Time histories are developed by fitting recorded seed time histories to the 5% damped target design spectra to meet the requirements of Clause 4.4.4 of CSA N289.3 and Section 5.2.3 of NEDO-33914 Revision 2.

Per the recommendations of NEDO-33914 Revision 2, seed time histories are selected from the NUREG/CR-6728 database of ground motion records. The selected seed time histories include records with different magnitude and distance bins that have spectral shapes reasonably consistent with the spectral shape of the design target spectrum over the frequency range of interest and characteristics that reasonably represent the earthquake motions expected at the site. Since only a limited number of records for moderate and larger magnitude earthquakes are available for the Central and Eastern United States in the NUREG/CR-6728 database, transformed records from the Western United States are used. The transformation of these time records is performed to modify the spectra to correspond to Central and Eastern United States site conditions while preserving the realistic phase and amplitude relationships of the original records. Based on the DNNP PSHA deaggregated seismic hazard results, the selection of seed time records considered multiple bins for rock seed time histories, including records from magnitude 6 to 7 earthquakes at distances of 10 to 50 km, and the magnitude 7+ earthquakes at 10 to 50 km, 50 to 100 km, and 100 to 200 km.

Table 3.3-5 provides details of the selected five sets of time history records used for the development of the design time histories for SSI analyses of DNNP BWRX-300 RB. The five selected time histories are all from the 1999 Chi-Chi Taiwan earthquake (magnitude 7.6) that had a reverse fault mechanism that is appropriate for eastern North America. These time history records had sampling rates ( $\Delta t$ ) of 0.005 seconds, with a Nyquist frequency of 100 Hz, and were typically longer duration recordings. Records from the shorter distances of 10 to 50 km and 50 to 100 km better matched the shape of the bounding ground motion response spectra once scaled to match the target spectrum at 100 Hz. The magnitude 7+ earthquakes at shorter distances than the scenario earthquakes (e.g., 10 to 50 km) are consistent with the target ground motion response spectra that represent an UB estimate of the seismic hazard. Smaller magnitude earthquakes were not selected because of a deficit of low frequency energy and the need for larger scaling factors. Table 3.3-5 provides the scaling factors applied to the time histories prior to spectral matching to better align the seed response spectrum shapes to the target spectra.

The spectral matching procedure is implemented for fitting the seed time histories to the 5% damped target spectra that retains the phase spectra of the seed time histories, preserving the relative phasing between horizontal and vertical components, as well as, preserving the non-stationarity and randomness characteristics. The modified time histories are checked as follows to ensure they meet the criteria specified in CSA N289.3, Clause 4.4 and ASCE/SEI 43, Section 2.4:

1. The 5% damped ARS of the modified seed time history are computed at a minimum of 100 points per frequency decade per CSA N289.3, Clause 4.4.4.3, uniformly spaced over the log frequency scale. The average of 5% damped ARS of the five Acceleration Time Histories (ATHs) are compared to the 5% damped target acceleration spectrum at each frequency point in the range of 0.1 Hz to 100 Hz to ensure that:
  - a. The average ARS does not fall below the target spectra by more than 10% at any frequency point
  - b. The average ARS does not fall below the target spectra at more than nine adjacent frequency points and 6% of the total number of points where the ARS is calculated satisfying the requirements of CSA N289.3, Clause 4.4.4.4.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

2. In accordance with Clause 4.4.4.5 of CSA N289.3, the power spectral density of the modified ground motion history is computed as described in ASCE/SEI 4, Section 2.6.2, and shown not to have significant gaps in energy at any frequency over this frequency range.
3. The total duration of time histories has to be no less than 15 seconds with minimum strong motion duration of 6 seconds per CSA N289.3, Clause 4.4.4.2 and long enough to provide an adequate representation of the Fourier components at low frequency.
4. Time histories used as input for the seismic response analyses have a strong motion duration, and ratios  $V/A$  and  $AD/V^2$  (where  $A$ ,  $V$ , and  $D$ , are the peak ground acceleration, velocity, and ground displacement, respectively) that are consistent with those of appropriate controlling events developed using the disaggregation data from in NK38-CORR-03611-0847339 (Reference 3.3-28)
5. The set of three modified ATHs representing the ground motion in the three orthogonal directions (two horizontal and one vertical) are statistically independent. Each pair of ground motion histories is considered statistically independent when the absolute value of their correlation coefficient does not exceed 0.16, satisfying the requirement of CSA N289.3, Clause 4.4.4.6.
6. The ATHs are baseline corrected to ensure the ground velocity converges to zero at the end of the earthquake record and maintains a zero-mean value over the time history duration.

Per recommendations of NEDO-33914 Revision 2, Section 5.2.3, the time step of the modified time histories is refined to 0.0025 seconds for the purposes of calculating high frequency in-structural responses, which exceeds the requirements of CSA N289.3, Clause 4.4.4.2.

Spectral matching of the seed time histories is completed using the time domain spectral matching procedure proposed by Lilhanand and Tseng (Reference 3.3-29) and later modified by Abrahamson (Reference 3.3-20) and Al Atik and Abrahamson (Reference 3.3-31). Figure 3.3-151, Figure 3.3-16, and Figure 3.3-17 present an example comparison of the original and spectrally matched time histories for the HWA026 records matched to the target rock design ground motion response spectrum. These plots demonstrate the non-stationary characteristics of the time histories are preserved. The most noticeable changes to the time histories are due to low frequency wavelets added at later portions of the time histories.

Response spectrum of the generated acceleration time histories are computed and compared to the appropriate target response spectra. A small scaling factor is applied to the time histories to increase the spectra and meet the design criteria. Finally, the cross-correlation coefficients, peak values, Arias Intensity, and Power Spectral Density function are computed for the spectrally matched time histories.

Figure 3.3-18 presents the normalized Arias Intensity, and the power spectral density function for the horizontal HWA026 components that are spectrally matched to the rock design ground motion response spectrum. Figure 3.3-19 presents the response spectra for spectrally matched horizontal and vertical components of record HWA026.

#### **3.3.1.1.5 Percentage of Critical Damping**

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, damping values assigned to the structures and components in the SSI analysis model are in accordance with provisions of CSA N289.3, Clause 6.6, and ASCE/SEI 43, Section 3.3.3. The damping ratio values specified in Table 4(a) of CSA N289.3, Table 3-1 of ASCE/SEI 43, and U.S. NRC RG 1.61 (Reference 3.3-32) are used to represent the dissipation of energy in different elements. Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, lower (Response Level 1) damping ratios are used for generating in-structure demands for qualification



of equipment and systems. The higher (Response Level 2) damping values can be used for development of seismic demands for structural design per ASCE/SEI 43, Section 3.3.3 and U.S. NRC RG 1.61, Section C.1.2, respectively.

The damping properties assigned to soil materials in the SSI analysis model take into account the stress-strain properties corresponding to the level of seismic input per requirements of CSA N289.3, Clause 6.6.3. Stiffness and damping properties of subgrade materials compatible to the strains generated by design level earthquake event are developed based on results of Approach 1 SRA in Subsection 3.3.1.1. The strain-compatible damping of the subgrade materials is limited to 15% in accordance with the recommendations of ASCE/SEI 4, Section C5.2 and the regulatory guidance of U.S. NRC RG 1.208, Appendix E.

Table 3.3-6 lists damping values used in the seismic analysis of structures and components. These damping values are applicable to all modes of a structure or component constructed of the same material.

Damping values for subsystems including piping and equipment are obtained using the procedures described in Subsection 3.3.1.3.

#### **3.3.1.1.6 Supporting Media for Seismic Category A Structures**

Consistent with regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the input subgrade properties for the site-specific SSI analysis of the BWRX-300 integrated RB structure are based on the geological, seismological, and geotechnical investigations and take into account the random nature and inherent uncertainties of soil material properties.

In accordance with the regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the SSI analysis uses at least three sets of subgrade profiles representing BE, UB, and Lower Bound (LB) estimates of the subgrade material properties. These profiles are representative of the as-built conditions at the DNNP site. The LB and UB shear wave velocities and damping reflect a minimum coefficient of variation of each layer properties of  $\pm 50\%$ . In accordance with CSA N289.3, Clause 5.2.3, the design uses an envelope of results from the SSI analysis of BE, LB and UB subgrade profiles to account for the variation and uncertainty in subgrade properties.

The effects of primary non-linearity of subgrade materials response are addressed by using dynamic stiffness and damping properties which are compatible to the free-field strains induced by an DBE level seismic event.

The strain-compatible subgrade dynamic properties for the DNNP soil materials are calculated in accordance with the requirements of CSA N289.3, Clause 5.2 and ASCE/SEI 4, Section 2.4. These properties are developed at strain levels consistent with the estimated site PBSRS based on the results of the probabilistic SRA presented in Subsection 3.3.1.1.2. The strain-compatible subgrade dynamic properties are developed using the approach described in Appendix B of the Screening Prioritization and Implementation Details document (Reference 3.3-19) as follows:

1. Strain-compatible shear wave velocity and damping ratios are obtained consistent with the 1E-04 and 1E-05 MAPE from the results of SRA of the BE-BE, LR-BE, and UR-BE randomized soil profiles discussed in Subsection 3.3.1.1.2.
2. The logarithmic mean and logarithmic standard deviation of the strain-compatible shear wave velocity and damping ratios at 1E-04 and 1E-05 MAPE are calculated for the considered cases at each soil layer. The results from different soil cases are combined using weight factors of 0.4, 0.3, and 0.3 for the BE-BE, LR-BE, and UR-BE base cases, respectively. The LR and UR kappa base cases (e.g., BE-LR and BE-UR) are not considered given their small effects on site response analysis results when compared to the alternative cases for shear wave velocity. The weighted logarithmic mean and logarithmic standard deviations of the strain-compatible

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

properties are calculated at 1E-04 and 1E-05 MAPE. The weighted average logarithmic mean and logarithmic standard deviation profiles for shear wave velocity and damping ratio at 1E-04 and 1E-05 MAPE are presented in Figure 3.3-20 through Figure 3.3-23, respectively.

3. The logarithmic mean and logarithmic standard deviation of shear wave velocity and damping ratio at strains that are compatible with the 100 Hz value of PBSRS are calculated by linear interpolation in the logarithmic space between those compatible with the 100 Hz values at 1E-04 and 1E-05 UHRS.
4. The exponential of the logarithmic mean profiles shear wave velocity profile calculated above is referred to as the median shear wave velocity and is selected as the 100 Hz BE shear wave velocity profile ( $V_{SBE}$ ). The LB and UB shear wave velocity profiles are calculated as the 16th and 84th percentiles, respectively, using the following equations:

$$V_{SLB} = \min \left\{ e^{\ln(V_{SBE}) - \sigma}, \frac{V_{SBE}}{\sqrt{1.5}} \right\}$$

$$V_{SUB} = \max \left\{ e^{\ln(V_{SBE}) + \sigma}, V_{SBE} \times \sqrt{1.5} \right\}$$

where  $\sigma$  is the logarithmic standard deviation and the terms  $V_{SBE} \times \sqrt{1.5}$  and  $V_{SBE}/\sqrt{1.5}$  reflect the minimum variation requirement of  $C_v = 0.5$  on the shear modulus as specified in CSA N289.3, Clause 5.2.3 to ensure that adequate uncertainty in the shear modulus of the soil profiles are included.

The 100 Hz strain-compatible LB, BE, and UB shear wave velocity profiles are presented in Figure 3.3-24.

5. The BE, LB, and UB profiles for damping ratio are calculated similar to step 4, except that no minimum variations of  $C_v = 0.5$  are used, and the damping ratios are limited to a maximum of 15%, based on the recommendations of ASCE/SEI 4, Section C5.2 and regulatory guidance of U.S. NRC RG 1.208, Appendix E. Consistent with non-linear behavior of soil layers, the 16th percentile of damping ratio profile is associated with the UB profile and the 84th percentile of damping ratios are associated with the LB profile. For the linear rock layers, a damping ratio logarithmic standard deviation of 0.6 is adopted. The 100 Hz strain-compatible LB, BE, and UB damping ratio profiles are presented in Figure 3.3-24.

$$D_{LB} = e^{\ln(D_{BE}) + \sigma}$$

$$D_{UB} = e^{\ln(D_{BE}) - \sigma}$$

6. The BE, LB, and UB profiles considering the interpolation at 1 Hz are established using the same approach described in Steps 3, 4 and 5 above.
7. The final BE profiles are calculated as the average of the BE profiles considering the 100Hz interpolated values and 1 Hz interpolated values. Similarly, the final LB and UB profiles are calculated as the average of their corresponding profiles for the 100 Hz and 1 Hz interpolations.
8. The compression wave velocity profiles ( $V_p$ ) are calculated using the final strain-compatible shear wave velocity profiles ( $V_s$ ) obtained in Step 7 and the Poisson's ratios ( $\nu$ ) recommended for each layer using the following equation. Note that below-ground water table, the minimum of the compression wave velocity of water (1,463 m/sec) and the compression wave velocity corresponding to a maximum Poisson's ratio of 0.48 is used.

The latter criterion is adopted to avoid numerical problems in subsequent SSI analysis of the structure.

$$V_P = V_S \sqrt{\frac{2(1-\nu)}{1-2\nu}}$$

9. The P-wave damping values used as input to the SSI analysis are limited to a maximum of 10% at large strains for soil layers above the ground water table.

The development of dynamic subgrade profiles considers the soils located below the nominal groundwater table to be fully saturated. The groundwater level at elevation of 85 m CGD corresponding to a depth of 3 m below the plant grade at elevation 88 m CGD is considered as noted in Subsection 3.5.2.2. Figure 3.3-25 presents the strain-compatible shear wave velocity, compression wave velocity and damping ratio profiles used for the bounding design seismic analyses of BWRX-300 Seismic Category A structures discussed in Subsection 3.3.1.2.

### **3.3.1.2 Seismic Analysis of Seismic Category A Structures**

This section discusses the seismic analysis of the Power Block Seismic Category A structures which consist of the RB, containment, and containment internal structures.

In accordance with CSA N289.3, Clause 6.2.3, the seismic demands for the design of the BWRX-300 Seismic Category A and Seismic Category B SSC are obtained from the seismic response analyses of the Seismic Category A structures that consider:

- Effects of interactions of the structures and the foundations with the surrounding subgrade
- Variation in the soil and structural parameters
- Hydrodynamic loads (mass and stiffness effects)
- Structure-Soil-Structure Interaction (SSSI) effects with the adjoining RWB, CB, TB, and Reactor Auxiliary Bay structures

Per Subsection 3.2.3, the BWRX-300 Seismic Category A and B SSC are hosted in the integrated RB structure, with the majority of them, including most of the Reactor Pressure Vessel (RPV) and the containment structure, being located below the plant grade elevation.

Because a significant part of the RB structure is located below grade, the interaction of the structure with the surrounding soil is a very important factor for the integrity of the RB structure, its seismic response, and the distribution of seismic stress demands.

In order to adequately account for the SSI and SSSI effects per guidance of NEDO-33914 Revision 2, Section 5.1, the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4, is implemented for the design of the integrated RB structure. Seismic structural stress demands are obtained directly from the results of SSI analyses of combined models that include 3-Dimensional (3-D) Finite Element (FE) representations of the integrated RB structure and the surrounding soil and Power Block structures. The surrounding subgrade is represented by layered half-space continuum with equivalent linear elastic stiffness properties and complex damping. Simplified FE models represent the dynamic properties of the surrounding Power Block structures and their foundations.

The methodology used for development of the 3-D integrated RB FE model is described in Subsection 3.3.1.2.2, and the SSI modeling assumptions are presented in Subsection 3.5.1.1.2.

#### **3.3.1.2.1 Seismic Analysis Method**

##### **One-Step Seismic Analysis Method**

Seismic demands for the design of Seismic Category A and B SSC are obtained from SSI analyses performed in accordance with the provisions of CSA N289.3, Clause 5.3, and ASCE/SEI 4, Section

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

5, following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, and U.S. NUREG-0800 (Reference 3.3-33), SRP 3.7.2.

The BWRX-300 one-step seismic SSI analysis approach provides demands for the seismic design and qualification of SSC for all frequencies of interest and adequately captures the effects of SSSI for the integrated RB with adjacent structures and foundations. The BWRX-300 seismic analysis approach follows the guidance of NEDO-33914 Revision 2, Section 5.0 to address current limitations in U.S. NUREG-0800 SRP 3.7.2 when capturing the effects of seismic interaction of the deeply embedded RB structure with adjacent structures through the subgrade, as identified in NUREG/CR-7193 (Reference 3.3-34), Section 1.5.11.

The seismic SSI analyses are performed using the sub-structuring method in CSA N289.3, Clause 5.3.5, and ASCE/SEI 4, Section 5.4 and the ACS SASSI (a system for analyses of soil-structure interaction, see Appendix 3B) computer program to calculate the seismic response of the RB SSI system. The SSI analysis model consists of the integrated RB structure, the surrounding subgrade and the excavated volume of the subgrade materials replaced by the embedded portion of the RB structure, near field backfill materials and the models representing the dynamic properties of the foundations and structures surrounding the RB.

The sub-structuring method allows the seismic response of the SSI system to be obtained by subdividing the problem into a series of simple subproblems that can be solved separately. Using the principle of superposition, the results of different sub-analyses are combined to obtain the final solution for the SSI problem. The solution for the seismic response of the BWRX-300 RB structure, is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.

The linear elastic SASSI analyses are performed on one-step structural models that accurately represent the geometry and dynamic properties of the integrated RB structure and its interaction with the subgrade. These structural models have a refined FE mesh that is identical to the mesh of the models used for the static analyses, and that can transmit the entire frequency range of interest for the seismic design of the RB SSC. These models assume isotropic elastic material properties of structural members and surrounding subgrade and neglect any non-linearity at the soil-structure contact interfaces.

The linear elastic assumption allows a set of design and sensitivity SASSI one-step approach analyses to be performed on refined RB structural models with a large number of interaction nodes. The superposition principle, which is applicable only for linear elastic analyses, allows the SASSI stress results obtained from different dynamic and static analyses to be combined with the results of static analyses in seismic design load combinations.

Far-field interaction nodes are established at the surface of each soil layer through the RB shaft embedment depth to capture the horizontal and vertical components of the far-field motion in the SSI model. The responses calculated from these far-field interaction nodes are used to monitor the propagation of the input control motion through the RB embedment depth.

To account for the non-linear response of subgrade materials, strain-compatible subgrade properties are used that are developed based on the results of equivalent linear probabilistic SRA as described in Subsection 3.3.1.1. The uncertainties related to variation of soil and rock properties are addressed in the design of RB SSC by using seismic demands calculated as an envelope of the results obtained from SSI analysis cases of BE, LB, and UB subgrade dynamic profiles.

Input ground motion ATHs are applied to the SASSI model at the RB foundation bottom elevation as vertically propagating coherent:

- Shear waves for horizontal components of the input motion
- Compression waves for the vertical component of the input motion

The horizontal control motion is applied to the SASSI model in a manner that is consistent with the 1-D wave propagation SRA approach discussed in Subsection 3.3.1.1.

As described in Subsection 3.3.1.1, five sets of three input motion ATHs are used as input for the SSI analyses to mitigate the uncertainty in the computed responses due to the phasing of the time history frequency components.

As described in Subsection 3.3.1.2.3, uncertainties related to variations of the input SSI parameters are addressed by results of sensitivity analyses following the recommendations in Section 5.3 of NEDO-33914 Revision 2.

### **Frequencies of Analysis**

Following the guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the frequency range considered in the seismic SSI analysis is based on the frequency content of the input ground motion, the soil properties, the building dynamic properties, including properties of the subsystems, and the response parameter of interest.

The solution for the response of the SSI system is obtained at a selected set of frequency points and then interpolated for other frequency points. The analysis is performed for a cut off frequency value established based on the largest value required by the following four criteria of ASCE/SEI 4, Section 5.3.5(b):

1. Twice the highest dominant frequency of the coupled soil-structure system or
2. The highest structural frequency of interest, or
3. The frequency at which the Fourier amplitude of input motion has passed its peak value and has reached 10% of the peak value, and
4. 20 Hz.

Criteria used to determine the highest dominant frequency and lower cutoff frequency values are described in Section 5.3.2 of NEDO-33914 Revision 2.

Sensitivity SSI analyses required to determine lower cutoff frequency values are performed for the stiffest UB subgrade profile that provides bounding responses at high frequencies.

The value of cutoff frequency determined by the criteria described above is used for the analysis of the UB subgrade profile. The analyses of the softer BE and LB profiles may use lower values for the cutoff frequency. In this case, it shall be demonstrated that the analysis of the UB profile provides responses that are bounding for frequencies higher than the cutoff frequencies used for the analyses of the softer subgrade profiles by comparing transfer function and 5% damped In-Structure Response Spectra (ISRS) results for responses at key locations within the building, selected as described in Subsection 3.3.1.2.5.

The frequencies of analysis are selected at sufficiently small frequency intervals. Transfer function amplitude results for responses at the key locations, selected as described in Subsection 3.3.1.2.5, are inspected to detect any numerical anomalies in the interpolated transfer functions (e.g., sharp narrow spikes) that can potentially affect the accuracy of results. If present, the effects of these anomalies in the interpolated transfer function results are evaluated using additional frequencies of analysis to ensure the anomalies in the transfer function interpolations do not affect the accuracy of the calculated responses.

Acceleration transfer functions and 5% damped ARS are also calculated for the response of SSI model free-field interaction nodes to check the amplitude and frequency content of the in-column free-field motion throughout the RB embedment depth.

#### **3.3.1.2.2 Procedures Used for Analytical Modeling**

SSI analyses of the integrated RB structure, which is primarily constructed of Steel Bricks™ as described in Subsection 3.5.1, are performed on 3-D FE models that meet the structural modeling requirements of CSA N289.3, Clauses 5.3.2 and 6.2, and ASCE/SEI 4, Section 3.

In addition to the integrated RB structures, simplified models of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and their foundations are included in the model to capture the SSSI effects in the RB seismic design.

#### **Dynamic Finite Element Modeling of Integrated RB Structure**

In accordance with requirements of Clause 6.10.4 of CSA N291, U.S. NUREG-0800, SRP 3.7.2, Subsection III.3.D, and ASCE/SEI 4, Section 3.4.2, the integrated RB structural FE model represents all mass expected to be present at the time of the earthquake including mass due to:

- Weight of the structure
- Weight of permanent equipment
- Mass equivalent to floor load of 2.4 kPa for miscellaneous dead weights such as minor equipment, piping, and raceways
- Weight of building elements not represented in the structural model (e.g., secondary members, siding partitions)
- Expected live load, not less than 50% of the live load specified for the design
- At least 25% of the specified design snow loads

The dynamic FE model also includes the inertia associated with the hydrodynamic effects of the fluids contained in various pools inside the RB and tanks in the RWB. The hydrodynamic effects that consist of the impulsive and convective (or sloshing) components are considered in accordance with the requirements of Clause 6.9 of CSA N289.3, Section 3.6.3 of ASCE/SEI 4, and Chapter 5 of ACI 350.3 (Reference 3.3-35). The hydrodynamic mass is included in the model by.

- Distributing the horizontal impulsive fluid mass over the pool and tank walls that are perpendicular to the direction of motion in accordance with the guidelines in ACI 350.3
- Lumping the entire vertical fluid mass on the pool slab or tank bottom.

The convective (sloshing) component of the hydrodynamic mass is not explicitly included in the global analysis model since its contribution is small and is associated with very low frequencies insignificant for the overall response. To account for the sloshing hydrodynamic effects, the design considers quasi-static sloshing pressure loads applied on the pool and tank walls in accordance with Section 9.4 of ASCE/SEI 4.

Beam and shell elements are used to adequately represent the configuration of all main structural members in the integrated RB. The FE model includes gross discontinuities such as large openings and member eccentricity. Thick shell elements are used to model the Steel Bricks™ shear walls, slabs, and mat foundation. 3-D beam elements are used to model the steel columns, beams, and trusses. The shell and beam elements are established at the centreline of the wall, slab, beam, column, and truss elements. Rigid beam and shell elements are used to model

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

member eccentricities and offsets or the section properties of the centreline modeled elements are appropriately adjusted to account for the effect of member offsets.

Local spring elements represent the stiffness of the connections between different structural members, such as the connections of the SCCV with the internal structures, RB walls and slabs that are designed to relief stresses due to thermal expansion.

Contact springs with stiffness properties appropriate to capture the interaction at the soil-structure interface connect the RB structural and subgrade FE models. The results obtained from the contact spring elements serve to:

- Calculate dynamic earth pressures on the below grade RB shaft exterior wall and basemat and
- Determine whether separation between RB shaft wall and soils occurs under DBE loading as discussed in Subsection 3.3.1.2.4.

The evaluation of effects of conditions at the contact interfaces with surrounding subgrade on the RB seismic response is discussed in Subsection 3.3.1.2.4.

The values of Young's modulus and Poisson's ratio representing the structural material stiffness properties are determined in accordance with the governing design codes in Section 3.5. BE stiffness properties are assigned to the concrete made structures in accordance with ASCE/SEI 4, Section 3.3.2.

The effective stiffness for analysis for the thick shell elements representing Steel Bricks™ members is determined in accordance with guidelines in ANSI/AISC N690 (Reference 3.3-36), Appendix N9, or equivalent guidelines that reflect the expected behavior of the structural components during the applicable loads. These guidelines are the same as those in NEDC-33926P (Reference 3.3-37), the licensing topical report providing design requirements for steel-plate composite containment vessel. The stiffness calculations account for the expected state of stress and level of cracking for different loading conditions during normal operation and accident conditions. An effective in-plane shear stiffness determined from ANSI/AISC N690 code Equation A- N9-12, may be used if seismic load is considered in combination with accident thermal loading.

ANSI/AISC N690, Equation A-N9-8 is used to calculate the effective flexural stiffness of Steel Bricks™ members based on the cracked transformed section, which accounts for stiffness from the steel faceplates as well as the cracked concrete infill. This equation is also used to account for reduction of flexural stiffness due to additional concrete cracking due to conditions related to accident thermal loading. The additional reduction in flexural stiffness due to accident thermal can be ignored for operating thermal conditions where thermal gradients are small and develop over longer periods of time.

For structural components whose behavior is controlled by membrane behavior, the effective stiffness for analysis for applicable loading conditions includes considerations to realistically represent the membrane stiffness calculated in accordance with industry accepted guidelines.

The effects of variation of structural stiffness and damping properties is considered in the modeling of the integrated RB structure to ensure accuracy of the calculated seismic responses and seismic demands. Section 5.3.5 in NEDO-33914 Revision 2 describes methods used and sensitivity analyses performed to evaluate possible amplifications of in-structure responses and load demands on the members due to the load redistribution effects.

The FE models used for seismic SSI analyses have a sufficiently refined mesh to be capable of transmitting the entire frequency range of interest for the seismic design of the RB SSC. In accordance with the requirements of ASCE/SEI 4, Section 5.3.4, the FE mesh is smaller than or

equal to one-fifth of the smallest wavelength transmitted through the soil model, i.e., the maximum mesh size:

$$d_{max} \leq \frac{V_s}{5 f_{cutoff}}$$

where:  $V_s$  is the shear wave velocity of the transmitting soil material; and

$f_{cutoff}$  is the cutoff frequency of analysis determined as described in Subsection 3.3.1.2.1

Consistent with requirements of CSA N289.3, Clause 5.3.4.4, the integrated RB FE model is sufficiently refined to ensure:

- Accuracy of SSI solution and ability to capture modes of vibrations up to frequencies that are important for the design
- SSI model can accurately transmit seismic waves with frequencies equal or higher than the cutoff frequency of analysis

Finer meshes are used around penetrations and openings that are larger than half of the wall or slab thickness. Meshes of major walls and slabs consists of at least four shell elements along the short direction and at least six shell elements along the long direction.

The lower boundary of the SSI model is established at a distance that is deeper than at least two times the depth of the RB embedment and at least three times the largest foundation dimension from the bottom of the slab in accordance with requirements of CSA N289.3, Clause 5.3.4.3.

### **Dynamic Modeling of Subsystems, Components and Equipment**

The dynamic properties of subsystems, components, and equipment are included in the integrated RB structural model based on the decoupling criteria of CSA N289.3, Clause 6.3, and ASCE/SEI 4, Section 3.7, depending on the ratios of the mass and first natural frequency of the subsystem, component, or equipment to those of the supporting structure. To capture the dynamic coupling effects of the RPV, the dynamic properties of the RPV and its components are represented by a Lumped Mass Stick (LMS) model capable of capturing all significant modes of the RPV seismic response. Procedures used to develop this LMS model are presented in Subsection 3.3.1.3. The RPV LMS model is connected to the RB structural model using local spring elements, representing the stiffness of the RPV support skirt and the horizontal stabilizers.

#### **3.3.1.2.3 Seismic SSI Analyses Results and Comparison of Seismic Responses**

##### **Key Seismic Responses**

Responses at key nodal locations are calculated to check the accuracy of the SSI analysis and to evaluate seismic responses and effects of variations of different SSI parameters. These key locations are selected based on the following criteria:

1. Nodes at intersections of main structural members (main structural walls) at ground and other major floor elevations to illustrate global responses that exclude possible local effects due to out-of-plane vibrations of slabs and walls, openings or connections with columns, beams or subsystem supports.
2. At least two roof nodes, one central and one corner node, to show all important modes of seismic response of structure including the effects of rocking and torsion.



3. At least two basemat nodes, one central and one corner node, to show the SSI effects on the translational as well as the rotational (rocking and torsion) responses of foundation.

The seismic demands on the below grade portion of the RB structure are affected by the deformations resulting from the response of the SSI system. Therefore, besides the in-structural responses, main stress demand components, such as in-plane shear force and vertical bending moment demands, are also compared to be able to gain a complete understanding of the effects of SSI parameters variations on the structural design. These comparisons are performed for the main below grade structural members at selected design cross-sections subjected to high seismic stress demands.

### **Seismic SSI Analyses Results**

Refer to Appendix 9B.B in Chapter 9B for results obtained from the Seismic SSI Analyses of BWRX-300 Seismic Category A structures.

#### **3.3.1.2.4 Seismic Soil-Structure Interaction Parameters**

The following are key requirements and approaches considered in the seismic SSI analyses to ensure the structural integrity and stability of the deeply embedded BWRX-300 RB structure throughout the life of the plant and to address specifics related to its design and construction.

#### **Implementation of ISG-017 Guidance**

BWRX-300 approaches for meeting U.S. NRC DC/COL-ISG-017 guidance and addressing current limitations in DC/COL-ISG-017 related to the seismic analysis of deeply embedded structures, as identified in NUREG/CR-7193, Section 1.5.8 are described in NEDO-33914 Revision 2, Section 5.3.4.

The intent of U.S. NRC DC/COL-ISG-017 is to ensure that the deterministic SSI analysis of the embedded RB structure uses ground motion inputs that are hazard consistent with the results of probabilistic SRA at the foundation bottom elevation and at ground surface.

The consistency between free-field motion at the bottom of the RB foundation used as input for the deterministic SSI analysis and probabilistic SRA is checked as described in Subsection 3.3.1.1, using the procedure described in Section 5.3.4.1 of NEDO-33914 Revision 2.

The augmented and smoothed horizontal and vertical 5% damped spectra presented in Figure 3.3-14 define the amplitude and frequency content of the SSI input control motion applied to the SSI model at the RB foundation bottom that is hazard consistent with the results of the probabilistic SRA described in Subsection 3.3.1.1.

#### **Coupling of Soil and Structures**

The seismic SSSI of the RB with the adjacent RWB, CB, TB, and Reactor Auxiliary Bay is explicitly considered in the seismic analysis and design.

Simple FE models representing the BE dynamic properties of the surrounding buildings and foundations are included in the integrated RB FE model used for the seismic SSI analysis. These simple models are sufficiently refined to capture all global modes of vibration of the RWB, CB, TB and Reactor Auxiliary Bay structures with significant (> 20%) modal mass participations in the three orthogonal directions.

Subsection 3.3.1.2.8 presents the approach for addressing the requirements related to the seismic interaction of the RB with the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and foundations.

#### **3.3.1.2.5 Effects of Parameter Variation on Responses**

This section covers the effects of concrete cracking, excavation support and backfill, groundwater variation, soil separation, non-vertically propagating seismic waves and soil secondary non-linearity on the seismic response and design of the BWRX-300 RB. The evaluations are performed in accordance with the requirements of ASCE/SEI 4, Section 5.1, following the guidelines of NEDO-33914 Revision 2, Section 5.3. They are based on comparisons of key in-structure responses, defined in Subsection 3.3.1.2.5, obtained from sensitivity SSI analyses as described below.

#### **Effects of Variation of Structural Stiffness and Damping Properties**

Effective structural stiffness and damping properties developed as discussed in Subsections 3.3.1.2.2 and 3.3.1.2.3 are assigned to the SSI model following the recommendations in Section 5.3.5 of NEDO-33914 Revision 2. Effective stiffness assigned to concrete members takes into account the level of stress in the concrete members due to the most critical seismic load combinations.

To address the effects of structural stiffness variations, sensitivity SSI analyses are performed on models representing lower structural stiffness properties corresponding to accident thermal and high intensity load conditions. Higher Response Level 2 damping properties may be used for the analysis of the model with LB structural stiffness.

These sensitivity analyses are performed for BE subgrade profile to evaluate the significance of the structural stiffness variations on the RB in-structure responses and redistribution of load demands on the structural members. The effects of structural stiffness variations are assessed by comparing key in-structure responses, defined in Subsection 3.3.1.2.5, of the two sensitivity analyses of models with reduced stiffness properties with results of the design basis analysis performed on the model with effective stiffness properties.

#### **Excavation Support and Backfill Effects**

Excavation support and backfill effects are to be addressed following the guidelines of NEDO-33914 Revision 2, Section 5.3.8. Sensitivity seismic SSI analyses are to be performed using BE properties of surrounding in-situ subgrade materials on a RB FE model that includes the excavation support structure and the fill concrete to assess their effect on the BWRX-300 RB seismic response. Shell and beam elements are to be used to represent the BE dynamic properties of the excavation support structure. Solid elements are to be used to represent BE, and the dynamic properties of concrete fill material. The geometry of the excavation support and the lean concrete are to be modeled based on the nominal dimensions obtained from excavation plan drawings. To address the uncertainties related to the modeling of friction at the RB shaft interfaces, the sensitivity SSI analyses are performed considering two bounding conditions:

- A. Fully bonded conditions assuming no slippage between the RB shaft and surrounding materials
- B. No-friction conditions assuming no friction resistance of RB shaft exterior walls

Results of these sensitivity analyses for key in-structure responses, defined in Subsection 3.3.1.2.5, are compared with the corresponding results of the design basis SSI analyses of FE model that excludes the excavation support and the fill concrete. If the comparisons show significant exceedances ( $> 10\%$ ) in the RB seismic response due to the interaction with the excavation support and fill concrete, the results of these sensitivity analyses are included in the RB seismic design basis.

### **Groundwater Variation Effects**

The potential effects of groundwater level variability on the seismic design of the BWRX-300 RB are addressed as described in Section 5.3.10 in NEDO-33914 Revision 2.

The seismic design of RB is based on analysis of SSI models that reflect fully saturated conditions for all soil materials located below the nominal groundwater elevation. The potential effects of groundwater level variability on the seismic design are addressed by comparing the seismic responses obtained from two sensitivity analyses of:

- A. Fully saturated soil profile with BE soil dynamic properties representative of accidental flood groundwater level
- B. Dry soil profile with BE soil dynamic properties representative of the extreme conditions when the groundwater is located below the RB foundation bottom elevation

Results of these two sensitivity analyses for key in-structure responses, defined in Subsection 3.3.1.2.5, are compared with the results of the design bases SSI analyses based on fully saturated soil profiles below the nominal groundwater elevation. If the comparisons show that the effects of groundwater variation significantly exceed (>10%) the design basis, the results of the two sensitivity analyses are included in the RB seismic design basis.

### **Soil Separation Effects**

The SSI analysis of the BWRX-300 RB addresses the uncertainties related to the inability of linear models used for the seismic design SSI analysis to explicitly represent the separation between the soil and the structure in accordance with the guidance of ASCE/SEI 4, Section 5.1.9(b).

The approach described in Section 5.3.9 of NEDO-33914 Revision 2 is followed to determine if the separation at soil-structure interfaces can have significant effect on the seismic response. A sensitivity SSI analysis is performed on a model where portions of the below grade shaft wall that may experience separation from the subgrade soil are assumed to remain unbonded for the total duration of the earthquake. The extent of soil separation is assessed by comparing the maximum lateral earth pressure calculated from the seismic SSI analysis of BE subgrade profile with a LB estimates of static earth pressures. The static lateral pressures calculated from static design SSI analysis with 1-g loading, described in Subsection 3.5.2.4, are reduced by 10% to account for uncertainties in calculation of soil unit weights and surcharge loads. The regions where the static lateral pressure is lower than the seismic lateral pressure are considered separated in the model used for the sensitivity analysis.

The key in-structure responses, defined in Subsection 3.3.1.2.5, and stress demands calculated from this sensitivity analysis are compared to the corresponding results of the SSI analysis of the model with BE properties representing fully bonded conditions. If the comparisons indicate that the seismic in-structure responses and stress demands from the fully separated model exceed those obtained from the SSI analysis of fully bonded models by more than 10%, the results of this sensitivity analysis are included in the RB seismic design basis.

### **Effects of Non-Vertically Propagating Seismic Waves**

The potential for non-vertically propagating seismic waves at the DNNP site is to be assessed following the guidelines in Section 5.3.3 of NEDO-33914 Revision 2 based on the geological and seismological conditions of the site. The available site information does not indicate presence of dipping soil and rock layers or local seismic sources that can result in significant non-vertical seismic wave propagation at the DNNP site

#### **3.3.1.2.6 Three Components of Design Ground Motion**

Earthquake motion is three-dimensional and seismic design takes into account the effects of three orthogonal components (two horizontal and one vertical) of the prescribed design earthquake.

The SSI analyses are performed separately for each of the three directional components of input ground motion using five sets of time histories per Subsection 3.3.1.1. For each set of time histories used as analysis input, the seismic response parameters obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response with either of the three methods permitted under ASCE/SEI 4, Section 4.2.2.

1. The time histories of responses due to the three earthquake components are combined algebraically on the time-step-by-time-step approach.
2. The maximum co-directional responses can be combined using the 100-40-40 method.
3. The maximum responses due to the three earthquake components can be combined using the Square-Root-of-the Sum of the Squares (SRSS) method.

The absolute sum method used in time domain may also be implemented (e.g., for calculations of seismic demands for foundation bearing pressure and stability evaluations) as a conservative alternative to performing the algebraic sum method for all possible combinations of the input motion directions.

#### **3.3.1.2.7 Development of In-Structure Responses**

ISRS and ATHs are developed from the seismic analysis to serve as input for the seismic design and evaluation of subsystems, components, and equipment.

##### **In-Structure Response Spectra**

The ISRS for the seismic design and evaluation of subsystem, components, and equipment are developed in accordance with the requirements of CSA N289.3, Clause 6.5.2.3 and ASCE/SEI 4, Section 6.2.

A set of ISRS are developed for required damping levels defining the amplitude and frequency content of in-structure design motion at different locations within the RB, in the two horizontal and the vertical directions for seismic qualification of substructures, systems, and components.

The ISRS for the design of subsystems for which dynamic properties are included in the global dynamic model using LMS models, are developed as an envelope of responses at the node locations where these LMS models are connected to the supporting structure provided that, per ASCE/SEI 4, Section 3.7.1(d), the LMS model adequately represents the major effects of interaction between the equipment and supporting structure.

The ISRS for the seismic design and evaluation of subsystems that are decoupled from the global model, and which location is known, are developed as an envelope of responses at the perimeter of the support footprint area to capture the effects of in-structure rotations. If the equipment or component is supported by flexible slabs or attached to flexible walls, ISRS are developed considering additional nodal responses that capture the local effects of out-of-plane vibrations of the supporting slab or wall.

If the LMS models are used to model the structure, substructure, or subsystem in the global dynamic model, the ISRS are developed as envelope of the responses of outrigger nodes located at the edges of the structure or subsystem.

In accordance with the requirements of ASCE/SEI 4, Section 6.2.1.1(a) and (b), the ISRS are developed from the calculated nodal in-structure responses by:

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

1. First combining in the time domain the three co-direction responses due to the three orthogonal components of seismic input motion as an algebraic sum at each time step and then calculating the ARS of the combined ATHs, or
2. Combining the co-directional ARS results obtained from the analysis with the three orthogonal components of seismic input motion using the SRSS method specified in Subsection 3.3.1.2.6.

The spectra are calculated for frequencies ranging from 0.1 Hz to the highest frequency of interest meeting the requirements specified in Table 2 of CSA N289.3. In addition, the ISRS are developed at small frequency intervals to ensure they are sufficiently close to the peak response frequencies of the supporting structure. To satisfy this requirement, the ISRS are calculated at 301 frequency points equally distributed on the logarithmic scale at the frequency range from 0.1 Hz to 100 Hz.

The ISRS are calculated as an envelope of the results from the seismic design basis SSI analysis of all subgrade profiles. In accordance with the requirements of Clause 6.5.2.3 of CSA N289.3, the peaks of the enveloping ARS are broadened by a minimum of +/-15% to address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology. The sharp valleys between peaks are filled to account for the uncertainties in subgrade properties.

### **In-Structure Acceleration Time Histories**

In accordance with the requirements of ASCE/SEI 4, Section 6.3, time histories used in the analysis of subsystems are obtained either:

- Directly from the results of the SSI analysis as time histories of nodal responses at reference of subsystem support locations; or
- By generating synthetic time histories compatible to multi-damping ISRS developed as described above.

When obtained directly from the SSI analysis results:

- Time histories of the co-directional in-structure responses due to the three components of the SSI analysis input motion are combined in the time domain
- Time histories are obtained from SSI analysis cases that are critical for the designed subsystem and include those obtained from BE soil case
- Time histories obtained from the BE soil case only can be modified by using time-shifting factors to address uncertainties related to the modeling of natural frequencies of supporting structure

### **Relative Displacements**

Relative Displacement between different support points of subsystems with multiple or distributed supports are evaluated using displacement time histories.

The time history of the relative displacements corresponding to each SSI analysis is obtained by algebraic calculation of the different displacement time histories at the support locations. Directional combination of the support displacement time histories is carried on a time-step-by-time-step basis. Maximum design relative displacements are calculated as an envelope of the maximum relative displacements obtained for each SSI analysis case.

#### **3.3.1.2.8 Seismic Interaction Evaluation**

Consistent with CNSC REGDOC-2.5.2, Section 7.13.1, the BWRX-300 design ensures the ability of the RWB, CB, TB, and Reactor Auxiliary Bay to prevent adverse interactions with the Seismic category A and B SSC during a DBE event.

To meet the interaction requirements in Subsection 3.2.3.1, evaluations are performed of the lateral load resisting system of the RWB, CB, TB, and Reactor Auxiliary Bay structures following the approach in NEDO-33914 Revision 2, Section 6.2. These evaluations are based on seismic responses of RWB, CB, TB, and Reactor Auxiliary Bay obtained from the SSSI analyses that incorporate the dynamic response of the RB and surrounding Power Block structures. As described in Subsection 3.3.1.2.2, models used in the SSI analyses of the RB include FE representations of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and foundations. The FE models of the RWB, CB, TB, and Reactor Auxiliary Bay are refined sufficiently to provide accurate stress demands on the major lateral load resisting structural members and accurate seismic displacements in the direction of the adjacent RB.

The seismic interaction evaluations consider limited permanent deformations (LS-C) structural response to calculate DBE demands for the main lateral load resisting structural members in accordance with the guidance of NEDO-33914 Revision 2, Section 6.2.

The stability of RWB, CB, TB, and Reactor Auxiliary Bay foundations is checked following criteria in Subsection 3.5.2.2 using demands calculated per Subsection 3.3.1.2.10. No reductions are applied to seismic driving force demands used for the stability evaluations to account for inelastic responses of these structures.

The resistance to sliding is calculated as summation of the effective cohesion and static frictional resistance between foundation and subgrade. The frictional resistance is based on the effective weight of the building and includes the buoyancy and seismic loads in the vertical direction. The lateral passive resistance of the foundation embedment soil is also considered, as applicable.

The overturning stability evaluation is performed for each orthogonal horizontal axis of the building using the overturning demands calculated per Subsection 3.3.1.2.10 and the restoring moments calculated using the effective weight of the building. The energy method described in BC-TOP-4A (Reference 3.3-38) can be used for overturning stability evaluation, where factors of safety against overturning are calculated by comparing the maximum kinetic energy driving the system to overturning during a seismic event with the potential energy required to prevent overturning of the structure and foundation. For this approach, the minimum overturning factor of safety of 1.25 is used, consistent with CSA N289.3.

The gaps between the RB and adjacent structures are evaluated per guidance in NEDO-33914 Revision 2, Section 6.2, to ensure no physical interaction between the RB structure and surrounding structures. The gaps are evaluated along the entire height of the adjacent structures considering construction tolerances, inelastic deformations, and possible differential settlements.

#### **3.3.1.2.9 Methods to Account for Torsion**

Considerations are given in the modeling of the integrated RB structure to represent the actual locations of the centre of masses and centres of rigidity of structural elements to account for torsional effects.

In accordance with the requirements of ASCE/SEI 4, Section 3.1, the seismic design of the RB structure also considers accidental torsion to account for:

- Non-vertically propagating seismic waves
- Rotational components of ground motion

- Possible distributions of structural mass and stiffness that differ from those represented in the 3-D FE model used for the seismic response analysis per the requirements in Clause 6.10 of CSA N289.3

Accidental torsional moment demands may be calculated at each floor level as the product of the story shear and 5% of the floor plan dimension perpendicular to the story shear direction. Alternatively, the horizontal shear force demands on all walls may be conservatively increased by 5% to account for the accident torsion.

#### **3.3.1.2.10 *Determination of Seismic Overturning Movement, Sliding Forces and Dynamic Bearing Pressures***

Contact spring elements installed in the SSI models at interfaces between the structure and the subgrade are used for calculation of seismic driving forces and overturning moments on the BWRX-300 foundations. As described in Subsection 3.3.1.2.6, time histories of the horizontal and vertical seismic forces in the three directions are calculated as the algebraic sum of the spring forces in the three directions at each step for all contact spring elements. Overturning moments about the two horizontal axes are calculated as the algebraic sum of the moments resulting from each spring force with respect to the foundation bottom centreline. Conservatively, the spring force results for calculation of seismic driving force demands may be combined using the absolute sum time domain method instead of using the algebraic sum method for all possible combinations of the input motion directions.

The seismic inertia forces and overturning moments for the foundation stability evaluations and seismic bearing pressure calculations are obtained from SSI models with higher (Response Level 2) structural damping values.

Seismic stability of the surface mounted foundations surrounding the RB are evaluated by calculating safety factors for seismic sliding and overturning stability for each time step. These safety factors are calculated for the total duration of each of the five sets of ATHs described in Subsection 3.3.1.1. The average value of the minimum safety factors obtained from the five sets of ATHs is used to demonstrate the seismic stability criteria described in Subsection 3.5.2.2 are met.

The seismic bearing pressure demands are also calculated in the time domain. Maximum bearing pressure values are calculated for the total duration of earthquake for each of the five sets of ATHs used as input for the SSI analysis discussed in Subsection 3.3.1.1. The dynamic bearing pressure demand under each foundation is defined as the average of the results obtained from the five sets of ATHs.

#### **3.3.1.3 *Seismic Analysis of Seismic Category A and B Subsystems***

This section applies to the Seismic Category A and Seismic Category B subsystems. Input motions for the qualification of these systems are usually in the form of floor response spectra or ATHs obtained from the primary system dynamic analysis discussed in Subsection 3.3.1.2. Input motions in terms of acceleration time histories are generally used. Dynamic qualification can be performed by analysis, testing, or a combination of both, or by the use of experience data. This section addresses the aspects related to analysis only.

##### **3.3.1.3.1 *Seismic Analysis Methods***

Seismic analysis of subsystems can be performed using one of the following methods:

- Time History Analysis
- Response Spectrum Analysis

- Static Coefficient

The time history and the response spectrum methods are utilized in the piping analysis as required. The procedure for multi-support excitation described in Subsection 3.3.1.3.9 is followed with both methods. When the multi-support Response Spectrum Method is used to calculate the dynamic response of the piping system, all multi-support response spectra components are simultaneously applied to each piping model for each load case.

The time history and Response Spectrum Methods are also utilized in the equipment analysis as required. When the equipment is supported at two or more points located at different elevations in the building, the response spectrum for the most severe single point of attachment is chosen as the design spectra. Alternatively, the multi-support excitation procedure described in Subsection 3.3.1.3.9 is used.

Vertical analyses of the RPV and internals are performed using in-structure responses obtained from the results of one-step analyses of the RB discussed in Subsection 3.3.1.2.

RPV and internal components such as fuel, guide tubes, and Control Rod Drive (CRD) System housing are included in the integrated RB model as discussed in Subsections 3.3.1.2 and 3.3.1.3.3. As a result, the evaluation of RPV internals components in the horizontal direction is performed using a Two-Step analysis approach, where seismic loads are applied to more detailed horizontal beam models of the RPV and internals. The first step of the Two-Step analysis consists, therefore, of obtaining ATHs or ISRS developed as described in Subsection 3.3.1.2 at the RPV/RB interface locations from the RB SSI analyses discussed in Subsection 3.3.1.2. The second step is a multi-support excitation time history analysis of the RPV, and internals subjected to the ATHs generated in the first step. The procedure for multi-support excitation time history analysis, as described in Subsection 3.3.1.3.9, is followed in the second step analysis of the RPV and internals.

### Time History Analysis

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M] \{ \ddot{u}(t) \} + [C] \{ \dot{u}(t) \} + [K] \{ u(t) \} = \{ P(t) \}$$

Where:

- $\{ u(t) \}$  = time dependent displacement of nonsupport points relative to the supports.
- $\{ \dot{u}(t) \}$  = time dependent velocity of nonsupport points relative to the supports.
- $\{ \ddot{u}(t) \}$  = time dependent acceleration of nonsupport points relative to the supports.
- $[M]$  = mass matrix.
- $[C]$  = damping matrix.
- $[K]$  = stiffness matrix.
- $\{ P(t) \}$  = time dependent applied force column vector.

The above equation can be solved by modal superposition or direct integration in the time domain.

Modal Superposition involves two steps. First, the characteristic equation corresponding to undamped, free vibration of the model is solved to obtain the eigenvalues, eigenvectors, and generalized masses. The system coupled equations are then decoupled via the eigenvector transformation matrix which is simply the matrix of eigenvectors written as columns. The equations are decoupled in the generalized coordinate system because of the orthogonality of the matrix of eigenvectors with respect to the “weighted” mass and stiffness matrices. The decoupled modal



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

equations are then solved independently to obtain the generalized coordinates. The physical solution is then given by the eigen transformation once the generalized coordinates are known.

The direct integration method involves the numerical integration of the simultaneous differential equations of equilibrium in their original form, without transformation to the generalized coordinates. For systems subjected to short duration, high frequency excitation (such as those due to LOCA acoustic, blast and jet loads), the direct integration method requires less computation and is recommended over the modal superposition method.

For the time domain solution, the numerical integration time step is sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency (or shortest period) of significance. This condition is satisfied if  $\Delta t$  is selected to limit the amplitude decay per cycle of free vibration of the highest significant mode to less than 20 percent. This corresponds to approximately 3.5 percent numerical damping for that highest significant mode. The integration time step for both the direct numerical integration of the system coupled equations of motion and the numerical integration of the  $n$  decoupled equations (Modal Superposition) satisfies the following requirement:

$$\Delta t \leq T_m/10$$

where  $\Delta t$  is the numerical integration time step magnitude and  $T_m$  is the period of the highest significant mode considered in the analysis or the reciprocal of the cutoff frequency in Hz as defined in Subsection 3.3.1.3.4.

### **Response Spectrum Analysis**

This method is used if only peak dynamic responses are required.

The response spectrum method is a modal superposition analysis in which only the peak values of the solution of the decoupled modal equations are obtained. The method is based on writing the solution of each decoupled modal equation in terms of the convolution integral. The major advantage of this form of solution is that for a given input motion the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor, it is possible to construct a curve which gives the maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity, consequently the maximum of the integral is called the spectral velocity.

For a subsystem analysis of a secondary system the input floor response spectra, obtained from a time history analysis of the primary system, is broadened  $\pm 15$  percent to account for modeling uncertainties in both the primary and secondary systems in accordance with ASCE/SEI 4, Section 6.2.3.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Subsection 3.3.1.2.

### **Static Coefficient**

The static coefficient method may be applied to certain equipment in lieu of the required dynamic analysis. Response loads are determined statically by multiplying the equipment mass by a static coefficient equal to 1.5 times the maximum spectral acceleration that corresponds to the first mode of the equipment. This coefficient is intended to account for the effect of both multi-frequency excitation and multi-mode response. This method is applicable only to equipment corresponding

to a simple column, beam, or frame type structure supported at a single point. Justification is required for applying this method or coefficient to equipment having configurations other than simple frame or beam type structures.

A factor of less than 1.5 may also be used if adequate justification is provided. For example, if the equipment is simple enough such that it behaves essentially as a single degree-of-freedom model and is greater than the seismic excitation frequency, the factor 1.0 can be used instead of 1.5.

If the fundamental frequency of the equipment is greater than the cutoff frequency but less than the Zero Period Acceleration (ZPA) frequency, the static coefficient can be taken as 1.5 times the peak spectral acceleration which occurs between the cutoff frequency and the ZPA frequency in the equipment input response spectra.

#### **3.3.1.3.2 Determination of Number of Earthquake Cycles**

The BWRX-300 Seismic Category A and Seismic Category B SSC are seismically qualified to withstand the effects of the DBE defined in Subsection 3.3.1.1. RW-IIa SSC are seismically qualified for one-half (1/2) of this DBE as stated in Table 3.3-1.

The determination of the number of earthquake cycles for subsystem analysis is in accordance with U.S. NUREG-0800, SRP 3.7.3.

#### **3.3.1.3.3 Procedures Used for Analytical Modeling**

The mathematical model for each Seismic Category A and B component to be analyzed is prepared to realistically reflect the dynamic characteristics of that component. Each component is discretized into a series of interconnected beam elements or finite elements. The node points are generally selected to coincide with the locations of large masses, such as at structure floors or at heavy equipment supports, and at all points corresponding to any significant change in physical geometry.

The number of mass node points in the model is sufficient if additional node points (independent of number) do not result in more than 10 percent increase in the responses in the frequency range below the cutoff frequency specified in Subsection 3.3.1.3.4.

The node point spacing is selected such that the maximum length  $L$  of the finite element between any two node points, in the direction of the stress wave propagation, satisfies the condition

$$L \leq \frac{\lambda}{4} = \frac{v}{4f} = \frac{vT}{4}$$

where:  $\lambda$  and  $v$  are the wavelength and wave velocity, respectively.

The frequency  $f$ , or period  $T$ , correspond to the cutoff frequency of Subsection 3.3.1.3.4.

#### **Modeling of Equipment**

For dynamic analysis, Seismic Category A and B equipment is represented by lumped mass systems which consist of discrete masses connected by weightless beam elements and/or by any other appropriate finite element representation. The criteria used to lump the masses are:

- A. The number of modes of dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than the cutoff frequency specified in Subsection 3.3.1.3.4.
- B. Mass is lumped at any point where a significant concentrated weight is located.

- C. For equipment with a free-end overhang span whose flexibility is significant compared to the centre span, a mass is lumped at the overhang span.
- D. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system.

### **Modeling of Piping Systems**

Mathematical models for Category A and B piping systems are constructed to realistically reflect the dynamic characteristics of the system. The continuous system is modeled as an assemblage of pipe elements (straight sections, elbows, and bends) supported by hangers and anchors, and restrained by pipe guides, struts, and snubbers. Pipe and fluid masses are lumped at the nodes and connected by the weightless elastic beam elements which reflect the physical properties of the corresponding piping segment. The mass node points are selected to coincide with the locations of large masses, such as valves, pumps, and motors, and with locations of significant geometry change. All concentrated weights on the piping system, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.3.1.3.4. The torsional effects of valve operators and other equipment with off-set centre of gravity with respect to the piping centreline are included in the analytical model. The pipe length between mass points is no greater than the length with a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.3.1.3.4 when calculated as a simply supported beam with uniformly distributed mass.

Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are excluded from the piping model of the main line in accordance with CSA N289.3.

All pipe guides and snubbers are modeled to produce representative stiffness to reduce model uncertainties. Snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The stiffness of the supporting structures is included in the analysis unless the supporting structure is shown to be rigid.

### **Modeling of Reactor Pressure Vessel and Internals**

Because of the significant dynamic interaction between the RB and RPV and internals, the latter are integrated into the RB model as discussed in Subsection 3.3.1.2.

The mathematical model of the RPV and internals consists of a LMS model connected by linear elastic members and 3D finite element models. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear.

To facilitate hydrodynamic mass calculations, mass points (e.g., representing the fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into two representative lengths. These lengths represent the longest and shortest housings to adequately represent the full range of frequency response of the housings. In order to reduce the complexity of the dynamic model, the light components (such as in-core guide tubes and housing, sparger, and their supply headers) are excluded from the RPV mathematical model. However, the dynamic response of selected components is determined from a subsystem analysis after the system response is found.

Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped inside the RPV vessel. Although the

dynamic coupling between the vertical hydrodynamic masses is not considered, the vertical hydrodynamic masses themselves are properly accounted for. Dynamic loads due to vertical motion are added to, or subtracted from, the static weight of component, whichever is more conservative.

The shroud support plate is modeled as a rigid link in the translational direction since it is loaded in its own plane during a horizontal dynamic event. The shroud support legs, and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

Due to the small clearances in the horizontal directions, the fuel assembly is adequately modeled as a linear system for subsystem and system analysis. In the vertical direction, the fuel assembly has the potential to lift off from its seat and a non-linear representation is required if the vertical applied and reaction forces are sufficient to cause fuel lift. Furthermore, the interface between the fuel channel and lower plate tie plate is not rigid and a non-linear model to account for slippage may be appropriate.

The weight of asymmetric secondary components, such as attached equipment, is uniformly redistributed around the node point circle. Asymmetric equipment is modeled using finite element or LMS methods.

#### **3.3.1.3.4 Basis of Selection of Frequencies**

The cutoff frequency selected in the time history and response spectrum analyses ensures that all significant modes are included in the superposition. Higher modes which cumulatively contribute less than 10% of the total system response are not considered in the superposition of the individual modal values.

The cutoff frequency for seismic and other dynamic loads follows Subsection 3.3.1.2. For seismic load, it is estimated that all modes up to 100 Hz are included.

For all other dynamic analysis, it is estimated that the cutoff frequency will be 100 Hz, as long as no more than 5 percent of the total strain energy of the system remains beyond this cutoff frequency.

Where practical, to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are approximately less than half or more than twice the dominant frequencies of the support structure. Moreover, in any case, the equipment is analyzed or tested or both to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

#### **3.3.1.3.5 Analysis Procedure for Damping**

##### **Damping of Primary Subsystems**

Primary Subsystems consist of the RPV and internals.

Damping values for seismic analysis of primary subsystems using the Modal Superposition are presented in Table 3.3-7. These damping values are in accordance with ASCE/SEI 43 and CSA N289.3.

$\alpha$ ,  $\beta$  –damping curves for the axis-symmetric finite element analysis of primary subsystems completed by Direct integration are defined per Table 3.3-8 and the following equation:

$$\lambda = \frac{\alpha}{2\omega} + \frac{\beta\omega}{2}$$

Damping values for dynamic loading beam analysis, performed by modal superposition, are identical to those for DBE provided in Table 3.3-7.

### **Damping of Secondary Subsystems**

Damping coefficients used in the seismic analysis of Seismic Category A and B piping, equipment, equipment supports and intermediate structures between subsystems are presented in Table 3.3-9.

Damping coefficients used for all other non-seismic loads are presented in Table 3.3-10.

These damping values are in accordance with ASCE/SEI 43 and CSA N289.3.

#### **3.3.1.3.6 Three Components of Design Ground Motion**

Applicable methods for spatial combination of responses due to each of the three input motion components are described in Subsection 3.3.1.2.

#### **3.3.1.3.7 Combination of Modal Responses**

Applicable methods for combination of modal responses are described in Subsection 3.3.1.3.1.

#### **3.3.1.3.8 Interaction of Other Subsystems with Seismic Category A and B SSC**

Non-Seismic Category systems are designed to be isolated from Seismic Category A and B systems by either a constraint or barrier or are remotely located with regard to the Seismic Category A and B systems.

If it is not feasible or practical to isolate the Seismic Category A or B system, adjacent Non-Seismic Category systems are analyzed according to the same seismic criteria as applicable to the Seismic Category A and B systems. Consistent with the approach used for evaluation of structures discussed in Subsection 3.3.1.2, limited inelastic deformation responses LS-C are considered for the seismic interaction evaluations of equipment by using inelastic absorption factors per ASCE/SEI 43, Section 8.2.2.2, and Table 8-1. For Non-Seismic Category systems attached to Seismic Category A and B systems, the dynamic effects of the Non-Seismic Category systems are simulated in the modeling of the Seismic Category A or B system. The attached Non-Seismic Category systems, up to the first anchor beyond the interface, are also designed in such a manner that during DBE level event it does not cause failure of the Seismic Category A or B system.

#### **3.3.1.3.9 Multiply Supported Equipment and Components with Distinct Inputs**

This section discusses the analytical method used for obtaining multi-support loadings and for dynamically analyzing Category A and B systems with multiple supports (or one support with many excitations), with different dynamic excitations. This analytical method is in accordance with CSA N289.3.

The time history Direct Integration, time history Modal Superposition and Response Spectrum Modal Superposition methods discussed in Subsection 3.3.1.3.1 can all be used in Multi-Support Excitation analysis. However, the mode superposition procedure described in Section 3.3.1.3.1 for an applied load vector is replaced with the corresponding mode superposition procedure for multi-support excitation analysis.

When using the time history method, the following methods are acceptable:

- A. The time histories corresponding to the envelopes of the ISRS for all attachment points in each of the three directions are applied at each attachment point simultaneously.
- B. The time histories corresponding to the envelopes of the ISRS for each attachment point in each of the three directions are applied at each corresponding attachment point simultaneously.

The above time history methods of analysis are performed such that primary (inertial) and secondary (static stresses due to differential displacements) are separated. The inertial forces are used for primary stress calculations. Secondary stresses are first computed for each natural mode of the supporting structures and for each excitation direction. The total secondary stress for triaxial excitation is then computed as the SRSS of the resultant secondary stresses for each excitation direction. The ASME BPVC Code Section III requires that the secondary stresses must be combined with the primary stress.

The inertia (primary) and displacement (secondary) stresses are dynamic in nature and their peak values are not expected to occur at the same time. Hence combination of the peak values of inertia stress and anchor displacement stress using the SRSS method is quite conservative. In addition, anchor movement effects are computed from static analyses in which the displacement are applied to produce the most conservative loads on the components.

Using the response spectrum method, support points response spectra are generated from support point acceleration time histories. In accordance with the requirements of Clause 6.5.2.3 of CSA N289.3,  $\pm 15$  percent peak broadening is applied to the spectra to account for the RB support structure modeling uncertainties. In general, using the SRSS method to combine modal responses is conservative since the maximum modal responses due to each component of multi-support excitation do not occur simultaneously. For certain "closely spaced" support with highly correlated support excitations, the SRSS superposition may yield unconservative responses. In this case, the modal responses of the "closely correlated" supports are combined algebraically first. Then, correlated sums are combined with the contributions for uncorrelated supports using the SRSS method.

#### **3.3.1.3.10 Use of Equivalent Vertical Static Factors**

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.3.1.3.1 are satisfied.

#### **3.3.1.3.11 Torsional Effects of Eccentric Masses**

Torsional effects of eccentric masses are considered in the modeling of subsystems as discussed in Subsection 3.3.1.3.3.

#### **3.3.1.3.12 Effects of Differential Building Movements**

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event.

Differential endpoint or restraint deflections cause forces and moments to be induced in the system. As discussed in Subsection 3.3.1.3.9, the stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

Refer to Subsection 3.3.1.2 for the methodology used to obtain differential displacements used in the evaluation of subsystems.

#### **3.3.1.4 Seismic Analysis of Other Subsystems**

Seismic demands for the evaluation of other subsystems are developed based on ISRS, ATHs and relative displacements calculated with the Response Level 1 structural damping values in accordance with CNSC REGDOC-2.5.2, Section 7.13.1. The use of models with higher (Response Level 2) damping values can be justified based on the level of stress response as applicable to these structures.

Per Clause 6.5.2 of CSA N289.3, the seismic input at support points for the dynamic analysis of decoupled subsystems are ISRS or time histories representing the in-structure design translational motion in the two horizontal and the vertical directions due to the three components of the input earthquake motion.

If the in-structure rotations are significant, rotational ISRS and ATHs are developed and used for the design of the decoupled subsystems. Relative displacements between different support points of subsystems with multiple or distributed supports are also considered in the evaluation.

#### **3.3.1.5 Seismic Instrumentation**

In accordance with the requirements in CNSC REGDOC-1.1.2 (Reference 3.3-39), Section 4.5.6, and CNSC REGDOC-2.5.2, Section 7.13.1, seismic instrumentation is used to monitor the seismic activity at the site for the lifecycle of the reactor facility, starting from commissioning, including outages, until fully decommissioned.

The design of BWRX-300 seismic instrumentation satisfies the more stringent requirements for large reactors in CSA N289.5, Clause 5 in addition to Clauses 1 to 3 and 8 to 10.

The handling of seismic instrumentation system data records is in accordance with requirements of Clause 10 of CSA N289.5. When required, the seismic instrumentation requirements of CSA N289.5 are augmented by the requirements of U.S. NRC RG 1.12 (Reference 3.3-40).

The required actions after an earthquake follow the provisions of CSA N289.1.

##### **3.3.1.5.1 Location and Description of Instrumentation**

#### **Free-Field Instrumentation**

In accordance with the requirements of Clause 5.2.2 of CSA N289.5, at least two triaxial accelerometers are installed outside of the structure-ground interaction influence of the Power Block, but as close as practicable to the reactor to monitor the free-field ground motion at the BWRX-300 site at the plant grade and close to the RB bottom elevations.

In accordance with U.S. NRC RG 1.12, Section C.1.2, because the deeply embedded RB is founded at a depth more than 12 m below finished grade elevation, installation of a second free-field downhole accelerometer is considered at the bottom of the RB foundation, below the free-field accelerometer at finished grade level.

#### **Structure and Equipment Instrumentation**

In accordance with the requirements of Clause 5.2.3.1.2 of CSA N289.5 and Section C.1.2 of U.S. NRC RG 1.12, triaxial accelerometers are installed at several locations inside the RB including:

- One at the top of the mat foundation
- One on the containment internal structure close to the reactor vessel
- One close to the top of the containment internal structure
- One close to the top of the containment structure
- One at the operating floor elevation

Also, in accordance with Clause 5.2.3.1.3 of CSA N289.5, three additional triaxial accelerometers are installed outside of the RB, either at locations of seismically qualified SSC or at other locations that are deemed important.

The specific locations for instrumentation are determined to obtain the most pertinent information consistent with the selected key locations in the RB model to enable easy comparison between

the measured and calculated in-structure responses. The sensors are installed such that occupational radiation exposures associated with their location, installation, and maintenance are maintained as low as is reasonably achievable.

Structure and equipment instrumentation stations recording are configured to be accessible for maintenance during full-power operation in compliance with the guidance of U.S. NRC RG 1.12. For sensors installed in inaccessible areas, provisions for data recording and an external remote alarm indicating actuation are provided.

### **Recording and Playback Equipment**

Recording and playback units are provided for multiple channel recording and playback of the triaxial accelerometer signals. Characteristics and installation requirements of the recording and playback equipment follow the guidelines in U.S. NRC RG 1.12.

Accelerometers can measure acceleration amplitudes of at least 2g in accordance with Clause 5.1.6.1 of CSA N289.5.

### **Power Sources**

In accordance with Clause 5.1.7.2 of CSA N289.5, a dedicated standby power source is provided for the seismic instrumentation. This backup power source can provide a minimum of 6 hours of continuous operation of any accelerometer or a minimum of 24 hours of continuous operation of any accelerograph in the event of failure of all external power sources.

The central unit of the seismic instrumentation system incorporates a self-contained seismically qualified standby power source dedicated for providing the system a minimum of 6 hours of continuous operation in the event of failure of all external power sources.

#### **3.3.1.5.2 Design and Installation**

In accordance with the requirements of Clause 8 of CSA N289.5, all components of the seismic instrumentation system and their supports are designed and installed to maintain their structural integrity, and to remain operational during and following a DBE. Accessibility for servicing and recalibration, anchorage and protection from adverse conditions that can affect their performance are also considered in the design.

Prior to the installation, the operational reliability of the seismic monitoring instrumentation is demonstrated, in accordance with Section C.4.7 of RG 1.12, by using prototype, environmental, vibratory, or historical test results.

#### **3.3.1.5.3 Maintenance and Testing**

Maintenance and testing of seismic instrumentation are defined in accordance with the requirements in Clause 9 of CSA N289.5, documented before the first facility startup, and updated as necessary following any modification to the system. All components of the seismic instrumentation system are maintained and tested to ensure that a maximum number of instruments are kept in-service during plant operation and shutdown.

The operability of each of the seismic instrumentations is demonstrated by performing channel checks every two weeks for the first three months of service after startup. After the initial three-month period and three consecutive successful checks, the channel checks are performed on a monthly basis. The channel calibrations are performed every 24 months or during each refueling outage. The channel functional test is performed every 6 months. At least once a year, the system is operated continuously on the standby power source to verify the required backup power availability per CSA N289.5, Clause 9.2.2.



The guidance of Appendix A to U.S. NRC RG 1.166 (Reference 3.3-41) is followed for instrumentation found to be out of service during an earthquake.

#### **3.3.1.5.4 Arrangements for Control Room Operator Notification**

In accordance with the guidance of U.S. NRC RG 1.12, Section C.4.13, the triaxial accelerograph system is triggered whenever a threshold free-field acceleration of not more than 0.01 g is exceeded for any of the three axes. A higher threshold value can be used if 0.01 g is impracticable due to the site geological or geotechnical conditions or the ambient noise at instrument locations.

Activation of the seismic trigger causes an audible and visual annunciation in the control rooms to alert the plant operator that a felt earthquake has occurred in accordance with Clause 5.1.3 of CSA N289.5. Authorities having jurisdiction as well as the local and regional emergency response agencies are advised of the plant status if an earthquake exceeds the threshold acceleration per Clause 6.5.4 of CSA N289.1.

#### **3.3.1.5.5 Comparison of Measured and Predicted Responses**

The appropriate response after a felt seismic event is determined by the level of shaking. In accordance with Clause 6.5.1 of CSA N289.1, the BWRX-300 post-seismic plant operation manual defines the response associated with each level of shaking. The required operator actions after a felt earthquake are in accordance with Clause 6.5.7 of CSA N289.1.

Per Clause 6.5.5 of CSA N289.5, an immediate shutdown of the plant is not mandatory if during and following an earthquake the plant continues successful operation. The plant is shut down if it is determined that the earthquake intensity exceeded the DBE or if there is evidence of damage impacting the safety systems.

In the event of a plant trip, all records pertaining to fuel and reactor internals systems are compared to the data that are recorded during a normal shutdown and/or previous plant trips. The intensity of the earthquake and any evidence of damage will dictate if a detailed inspection is required or if a restart is allowed. Prior to startup, the availability of all safety class SSC is confirmed to ensure they can perform their intended functions.

#### **Immediate Response Following a Seismic Event**

If the plant remains online following a seismic event, the immediate response is to stabilize the plant in accordance with Clause 6.5.7.1.1 of CSA N289.1 by:

- Testing all systems required to perform nuclear safety functions
- Initiating inspections performed in accordance with the provisions of ANSI/ANS-2.23 (Reference 3.3-42) to assess the intensity of the seismic events and the effects on essential systems

Recorded earthquake data from the seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination of whether the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event.

#### **Seismic Design Basis Exceedance**

Following a seismic event, records of free-field ground motion and in-structure responses are reviewed in accordance with Clause 6.5.6.1 of CSA N289.1.

Cumulative absolute velocity calculated in accordance with Section 6.4.1 of ANSI/ANS-2.23 and peak ground velocity are generated from all free-field ground motion to be used as damage

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

indicators. Damage criteria for Heavy industrial SSC in Section 6.5.6.2.1 of CSA N289.1 are also considered to help determine seismic design basis exceedance.

The DBE is considered exceeded when the measured free-field motion in any of the three directions (two horizontal and one vertical) exceeds the following limits:

1. Response spectrum limit that is exceeded if:
  - a. At frequencies between 2 and 10 Hz, the recorded response spectral accelerations of 5% damping exceed the corresponding DBE design acceleration response spectrum or 0.2 g, whichever is greater or
  - b. At frequencies between 1 and 2 Hz, the recorded response spectral velocities of 5% damping exceed DBE velocity response spectrum or 152 mm/sec, whichever is greater
2. Cumulative absolute velocity limit that is exceeded if the cumulative absolute velocity value calculated in accordance with Clause 6.5.6.1 of CSA N289.1 is greater than 0.16 g-s, or the peak ground velocity is greater than 50 mm/s.

The DBE exceedance is checked for measurements taken from the free-field plant grade accelerometers and downhole accelerometers using the corresponding design response spectra defining the DBE ground motion at the plant grade and RB foundation bottom elevations.

In addition to the criteria above, the following is also used to determine DBE exceedance:

- The inspection of the seismically qualified SSC shows evidence of overstressing, large displacement, yielded supports, etc.
- If the data collected from the monitoring instruments installed at different elevations in the plant exceed the DBE response parameters at the corresponding locations

#### **Required Pre-Shutdown Earthquake Actions**

Prior to the shutdown, the availability of safety class systems required for shutdown and the availability and integrity of the containment system are confirmed by performing pre-shutdown checks in accordance with the provisions of CSA N289.1, Clause 6.5.7.2.

#### **Post-Shutdown Earthquake Response Actions**

While the plant is shut down, a detailed inspection and evaluations are performed to assess the state of the plant in accordance with the provisions of CSA N289.1, Clause 6.5.7.3.

Post-shutdown actions include:

- Focused inspections of a preselected set of SSC that are representative of a broad cross section of equipment and structures in nuclear and conventional power plants
- Expanded inspections if damage is found in focused inspections
- Further graded inspections, tests, and analyses that are guided by the damage and earthquake levels

Focused inspections include detailed, visual inspections and tests of a preselected sample of representative structures and equipment, selected to sample all types of safety class and SCN SSC that are considered most likely to be damaged due to earthquake shaking. SCN SSC that experience has shown to be of low seismic capacity to serve as earthquake damage indicators are also included in the focused inspections.

Expanded inspections and tests are performed if significant physical or functional damage is found during the focused post-shutdown inspections. The expanded inspections include all accessible safety class equipment and structures as well as non-safety-class balance-of plant equipment that is important to safe operation of the plant. Expanded inspections and tests may not be performed if the damage observed as part of the focused inspections is isolated to a specific class of SSC and if the cause of the damage is attributable to a specific design or installation deficiency, such as lack of equipment anchorage, improper installation of expansion bolts, etc. In this case, the design or installation deficiency is corrected for all SSC in the classes involved, and inspections of other undamaged classes may not need to be expanded.

If damage to safety class SSC is observed, the reactor vessel is opened, and reactor vessel internals and fuel are inspected using methods normally employed for in-service inspections.

If the DBE is reached, the plant restart is only allowed after ensuring that the allowable design stresses of seismically qualified SSC are not exceeded.

Results of post-shutdown inspections and tests are documented and reported to the authorities having jurisdiction. Results of inspections are compared with results of previous baseline inspections.

### **3.3.2 Extreme Weather Conditions**

This section presents the design basis weather conditions considered in the design of the BWRX-300 SSC for the bounding extreme meteorological hazards identified in Chapter 2, Section 2.6.

#### **3.3.2.1 Temperature and Humidity**

The extreme temperatures and humidity levels specified in Chapter 2, Table 2.6-1 are considered in the BWRX-300 design in accordance with CNSC REGDOC-2.5.2, Sections 7.4.2 and 7.15.1. Conservative safety margins are considered in the evaluations and design of SSC to ensure their availability and efficiency under extreme temperature and humidity conditions.

#### **3.3.2.2 Rain**

Rain load is considered in the design of the BWRX-300 building structures.

The RB roof is designed to minimize or eliminate rain loading in accordance with U.S. NRC RG 1.102 (Reference 3.3-43), regulatory position 3, considering rain intensity and duration (PMP) values listed in Chapter 2, Table 2.6-1.

Design for rain loading on the RWB roof is performed in accordance with CSA N291 Clause 6.2, considering PMP values specified in Chapter 2, Table 2.6-1.

The design of the remaining Power Block roofs to minimize and evaluate the potential of ponding follows the guidance in the NBC, Section 4.1.6.4.

#### **3.3.2.3 Snow and Ice**

The RB structure is designed using ground snow loads for normal and extreme winter precipitation events of 2.5 kPa and 5.0 kPa, respectively. These loads envelop those used in the design of the nearby Darlington Nuclear Generating Station listed in Chapter 2, Subsection 2.6.9. For the RB structure, ground snow loads are converted to roof snow loading in accordance with the methodology specified in the ASCE/SEI 7 (Reference 3.3-44) referenced in U.S. NRC DC/COL-ISG-7 (Reference 3.3-45).

For the RB structure, the normal roof snow load is considered as a normal live load for all normal operating load combinations considered in the design. The extreme roof snow load is considered as an extreme load for the extreme environmental combinations (See Chapter 9B, Table 9B-4), without concurrent seismic or tornado loads.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

For the RWB design, snow load (including snow drifting conditions, as applicable) is computed in accordance with the methodology specified in CSA N291, Clause 6.3 and NBC, and based on 100 years occurrence specified in Chapter 2, Table 2.6-1.

For the design of other Non-Seismic Category Power Block structures, the design snow load is determined in accordance with the methodology specified in NBC considering 50 years recurrence. The Importance Factor for Snow,  $I_s$ , assigned to these structures is based on Table 4.1.6.2-A of NBC for Post-Disaster importance category.

#### **3.3.2.4 Wind**

In accordance with REGODOC-2.5.2, Section 7.15.1, wind loads are considered in the design of the BWRX-300 building structures and components.

Site-specific wind speeds for the RB structure are translated into structural loading in accordance with the methodology specified in ANSI/AISC N690. The RB is designed as an ASCE/SEI 7 (referenced in ANSI/AISC N690), Risk Category IV structure (3000-year return period), for severe wind load of 257.5 km/h with 3-second gust basic wind speed that is bounding the site-specific design basis wind speed values in Chapter 2, Table 2.6-1.

Wind loads for the design of the RWB are determined in accordance with the methodology specified in CSA N291, Clause 6.3 and NBC, and based on 100 years return period wind pressure specified in Chapter 2, Table 2.6-1.

Wind loads for the design of other Non-Seismic Category Power Block structures are determined in accordance with the methodology specified in the NBC, Section 4.1.7. The reference wind speed is based on 50-year return period one-hour mean reference design wind. The Importance Factor for Wind,  $I_w$ , assigned to these structures is based on Table 4.1.7.3 of NBC for Post-Disaster importance category.

#### **3.3.2.5 Tornado**

In accordance with CNSC REGDOC-2.5.2, Section 7.15.1, tornado loads are considered in the design of BWRX-300 building structures and components based on their pertinent Seismic Category listed in Table 3.3-1.

Tornado loads included in the design of the Seismic Category A RB structure include:

- Tornado wind pressures
- Differential pressure loads due to rapid atmospheric pressure change
- Tornado-generated missile impact

The design input tornado wind parameters and tornado missile spectrum applicable to the Seismic Category A RB structure are provided in Chapter 9B, Table 9B.9-2 and Table 9B.9-3. These parameters are based on Region I values from U.S. NRC RG 1.76 (Reference 3.3-47). These values bound the DNNP site-specific parameters listed in Chapter 2, Table 2.6-5, and Table 2.6-6.

The RW-IIa RWB which houses rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes is designed for the site-specific tornado wind and missile spectrum modified per the requirements of Table 2 of RG 1.143.

The RWB, CB, TB, and Reactor Auxiliary Bay are evaluated for the design basis tornado wind loads applicable for the RB so that their interaction with the RB does not adversely affect the ability of the Seismic Category A and B SSC to perform their safety functions. The interaction evaluation follows the guidance of NEDO-33914 Revision 2, Section 6.3.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The structural integrity of the CB is maintained in the event of a design basis tornado missile to allow egress of operators to the Secondary Control Room (SCR) in the RB and to ensure availability of SSC providing post-disaster mitigation functions. For the special hardening provisions considered in the design of the CB, refer to Chapter 9B, Section 9B.3.2.2.

For a discussion of tornado dampers used to protect the Heating, Ventilation and Air Conditioning (HVAC) openings in the RB and CB to improve their survivability under tornado, refer to Chapter 9A, Section 9A.5.

The procedures for transforming tornado wind speed into pressure-induced forces to apply to structures and the distribution across the structures are based on BC-TOP-3-A (Reference 3.3-46). U.S. NRC RG 1.76 provides guidance to determine the pressure drop and rate of pressure drop caused by the passage of a tornado.

Missiles created as a result of components and cladding failing during a tornado wind event are considered enveloped by the design basis missile spectrum considered for the RB.

### **3.3.2.6 Hurricanes**

Hurricanes at the DNNP site are considered bounded by tornado loads discussed in Subsection 3.3.2.5.

### **3.3.2.7 Lightning**

Complying with Section 7.4.2 of CNSC REGDOC-2.5.2, grounding and lightning protection systems are used to protect structures, transformers and equipment against lightning induced surges as described in Chapter 8, Section 8.6.

Protection measures against fires and electromagnetic compatibility issues that could affect the functionality of electrical systems as a result of lightning are addressed in Subsections 3.3.6 and 3.3.7.1.

### **3.3.2.8 Extreme Wind Interaction**

As described in Subsection 3.3.2.5, evaluations are performed to ensure that there is no adverse interaction between the RWB, CB, TB and Reactor Auxiliary Bay and the RB under design basis tornado wind loads applicable for the RB.

### **3.3.3 Extreme Hydrological Conditions**

Potential sources of external floods considered in the BWRX-300 design are discussed in Chapter 2, Subsection 2.5.3.

To conform with Section 7.4.2 of CNSC REGDOC-2.5.2 and in accordance with U.S. NRC RG 1.102, Seismic Category A and RW-IIa structures are designed to include protective features that are used to mitigate or eliminate the adverse consequences of flooding due to external sources.

Conforming with CNSC REGDOC-2.5.2, Section 7.15.1, the integrated RB structure is designed to withstand the maximum external flood and groundwater levels specified in Chapter 2, Section 2.5.3.1.

Protection measures considered for the integrated RB structure against underground water includes the use of:

1. Hydrostatic and hydrodynamic loads to design walls below flood level in conformance with CNSC REGDOC-2.5.2, Section 7.15.1
2. Suitable provisions to ensure water tightness of external surfaces and penetrations below design basis maximum flood and groundwater levels

3. No exterior access openings below grade

In accordance with U.S. NRC RG 1.143, the RWB is designed for one-half of the Probable Maximum Flood (PMF) listed in Chapter 2, Subsection 2.5.3.1.

Because plant grade is above design flood level, the Power Block structures remain accessible during postulated flood events. Thus, no emergency actions are required due to flooding to ensure the safe operation of the BWRX-300 plant.

**3.3.3.1 Analysis Procedure**

The BWRX-300 RB is analyzed and designed to withstand the effects of the maximum external flood and highest groundwater levels specified for the plant. The maximum flood and highest groundwater levels listed in Chapter 2, Subsection 2.5.3.1 are considered in defining the input design parameters for the structural design to account for flood and groundwater loadings.

Because the flood level at the DNNP site is below the finished grade level, only hydrostatic effects are considered in the analysis and design of structures, while dynamic phenomena associated with a flooding event, such as currents, wind waves, and their hydrodynamic effects are not considered. The hydrostatic pressure associated with the design flood level or with the design groundwater level is considered as a structural load on the basemat and basement walls for structural design in accordance with CNSC REGDOC-2.5.2, Sections 7.4.2 and 7.15.1. Uplift or floating of structures is considered and the total buoyancy force is based on the hydrostatic pressure due to the design flood level, excluding wave action, or the design groundwater level. The lateral, overturning and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, are also considered in the structural design of these elements.

**3.3.4 Aircraft Crash**

This section discusses non-malevolent, general aviation crashes in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.4.2. For robustness against malevolent acts, including aircraft crashes, refer to Subsection 3.3.7.4.

Small aircraft crashes are considered in the BWRX-300 design but are screened out per Chapter 2, Subsection 2.2.3.1. The design considers these aircraft crashes as missiles bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

To mitigate their potential of equipment damage and fire impacts, the design of the BWRX-300 Seismic Category A structures addresses penetration resistance of buildings and considers physical separation of redundant or backup equipment, where applicable.

**3.3.5 Missiles**

**3.3.5.1 Missiles Generated by Extreme Winds**

Refer to Subsection 3.3.2.5 for details.

**3.3.5.2 Site Proximity Missiles (Except Aircraft)**

The design considers site proximity missiles to be bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

Due to the distance between the sites, the maximum turbine missile from the existing Darlington site does not impact the DNNP site.

**3.3.5.3 Structures, Systems and Components to be Protected from Externally Generated Missiles**

Seismic Category A, RW-IIa, and portions of the TB and CB structures are designed to withstand the effects of externally generated missiles. For Seismic Category A SSC, the tornado wind

characteristics and tornado missile spectra considered in the design are listed in Chapter 2, Table 2.6-3 and Table 2.6-4. Tornado wind and tornado missile spectra design input values considered in the design of the RWB are listed in Table 2 of RG 1.143.

The response determination methodology due to missile impact loading on the RB structure, consisting of Steel Bricks™ modules, is in accordance with ANSI/AISC N690, Appendix N9.1, Section 6c.

The response determination methodology due to missile impact loading on the RWB and portions of the TB and CB is in accordance with CSA N291, Annex A.

#### **3.3.5.4 Barrier Design Procedures**

In accordance with CSA N291, Clause A.5, barrier design for impact loads satisfies the criteria for local and overall effect. The procedures for designing barriers to withstand the effects of missile impacts are per U.S. NUREG-0800, SRP 3.5.3.

##### **3.3.5.4.1 Local Damage Prediction**

The prediction for local damage in the impact area depends on the basic material of construction of the barrier.

#### **Concrete Barriers**

Sufficient thickness of concrete is provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact.

Per CSA N291, Clause A.5.2.3, empirical formulas are applicable over a limited range of missile and target parameters.

Required concrete barrier thicknesses are determined in accordance with U.S. NUREG-0800, SRP 3.5.3 and are in no case less than those of Region I listed in Table 1 of U.S. NUREG-0800, SRP 3.5.3. In accordance with CSA N291, Clause A.5.2.4, the required barrier or wall thickness to prevent perforation is at least 20% greater than the calculated thickness from the applicable empirical formulas. Also, the required barrier or wall thickness to mitigate missile penetration is at least 50% greater than the calculated thickness from the applicable empirical formula.

#### **Steel Barriers**

Steel barrier thicknesses are determined using the Stanford equation (Reference 3.3-48) in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

#### **Composite Sections**

Composite section barriers are utilized in the BWRX-300 for missile protection when the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element for prediction of local damage in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

##### **3.3.5.4.2 Overall Damage Prediction**

The BWRX-300 design for impactive loads satisfies the criteria for the overall effect of Clause A.5.3 of CSA N291. Dynamic effects of impactive loads are evaluated by dynamic analysis in accordance with Clause A.4.1.1 of CSA N291 or the equivalent static load approach mentioned in Clause A.4.1.2 of CSA N291.

#### **3.3.5.4.3 External Doors**

The RB external doors are designed to resist tornado missiles unless shielded by external stair towers or elevator shafts. External stair towers or elevator shafts credited for shielding are evaluated for tornado missiles.

#### **3.3.6 External Fires, Explosions and Toxic Gases**

In line with requirements of CNSC REGDOC-2.5.2, Section 7.4.2, damages due to fires, explosions, and release of toxic gases as a result of transportation and industrial accidents at or near the DNNP site are considered in the BWRX-300 design. The following subsections provide information on measures considered to protect and mitigate the effects of:

- External fires – Subsection 3.3.6.1
- Explosions – Subsection 3.3.6.2
- Release of toxic gases – Subsection 3.3.6.3

##### **3.3.6.1 External Fires**

Per Chapter 2, Subsections 2.2.3, 2.2.4, 2.4.1 and 2.6.10, sources of external fires at the DNNP site include fireballs as a result of a rail transportation accident, forest fires, lightning and accidental fires in on-site storage areas of hydrogen, liquid waste or fuel oil. As mentioned in Chapter 2, Subsection 2.2.4, the risk of fire due to pipeline ruptures close to the DNNP site is negligible and is therefore not considered in the design.

Chapter 9A, Section 9A.6 describes the BWRX-300 fire protection systems implemented to resist and mitigate the effects of external fires. Buildings and structures within the protected area are supplied fire water from redundant loops by two fire water storage tanks (See Chapter 9A, Section 9A.6.6) providing suction to fire pumps located in a Fire Pump Enclosure structure (See Chapter 9B, Section 9B.3.6).

Figure A1.4-1 in Appendix A of Chapter 1 shows the location of the fire water storage tanks and Fire Pump Enclosure at the DNNP site.

Protection measures against the release of toxic gases as a result of external fires are discussed in Subsection 3.3.6.3.

##### **3.3.6.2 Explosions**

The RB structure is designed to withstand impulsive and impactive loads as discussed in Subsection 3.5.5.4.

##### **3.3.6.3 Release of Toxic Gases**

On-site activities that could result in release of toxic gases that could impact the safe operation of the BWRX-300 DNNP are summarized in Chapter 2, Section 2.4. External sources of toxic gases and chemicals are discussed in Chapter 2, Section 2.2.

Mitigation measures considered in the design of MCR/SCR are referenced in Chapter 6, Section 6.4.

#### **3.3.7 Other External Hazards**

##### **3.3.7.1 Electromagnetic Interference**

Protection against electromagnetic interference caused by lightning, high-voltage transmission lines at DNGS and telecommunication towers (See Chapter 2, Subsection 2.2.9) is provided through the use of appropriate shielding and qualification of equipment.



Safety Class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

For a description of plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements, refer to Chapter 8, Section 8.6.

#### **3.3.7.2 Biological Phenomena**

In accordance with CNSC REGDOC-2.5.2, Section 7.4.2, the Pumphouse/forebay structure is designed to prevent clogging by algae and exceptional quantities of fish and to stop them from entering the cooling systems. Measures considered to mitigate the effects of such clogging include locating the intake tunnel and lakebed intake structure at an adequate depth in the lake and the installation of traveling water screens to prevent intake of biofouling material as described in Chapter 9B, Subsection 9B.3.5.

As shown in Chapter 1, Appendix A, Figure A1.4-1, the BWRX-300 protected area is fenced which, in turn, prevents entry of large animals into the plant.

Screens or equivalent engineered features are also provided to prevent blockage of outside air intakes by non-human biota.

#### **3.3.7.3 Collisions of Floating Bodies and Frazil Ice with Water Intakes**

To satisfy requirements in CNSC REGDOC-2.5.2, Section 7.4.2, the design of the intake structure includes measures to mitigate the potential risk of blockage by frazil ice accumulations and physical damages as a result of a marine accident.

Measures considered to preclude blockage by frazil ice include a proper design of the Circulating Water System (CWS) recirculation line to prevent the formation of frazil ice in the forebay. Refer to Chapter 10, Section 10.8 for information related to the CWS.

To prevent marine transportation accidents, a restricted zone is established around the BWRX-300 lakebed intake structure and discharge diffusers to stop commercial ships from approaching offshore structures as stated in Chapter 2, Subsection 2.2.3.4.

#### **3.3.7.4 Robustness Against Malevolent Acts**

The BWRX-300 design provides robust physical features for the protection against malevolent actions found in the Design Basis Threats (DBTs) and Beyond Design Basis Threats (BDBTs). This results in the following fundamental capabilities remaining available after malevolent actions intended to cause substantial radiological releases:

- Ability to shut down the reactor and maintain sub-criticality
- Ability to cool irradiated fuel, both in the core and in the fuel pool
- Ability to limit or prevent the release of radioactivity affecting public health and safety

The ultimate gauge of success of the above three key functions is the prevention of radioactive releases that impact the health and safety of the public.

The BWRX-300 development has included a security by design approach from the early stages of design that uses sound engineering principles to demonstrate that, within an acceptable margin of confidence, sufficient capabilities are available to perform the above functions over a wide range of threats. This approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter.

The following are examples of features that enhance protection against malevolent actions:

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Much of the RB structure, including the portion housing the RPV, is embedded underground, thereby naturally limiting access pathways.
- The number of entrances to the RB are minimized while maintaining emergency exits for personnel safety.

The BWRX-300 Security Annex further describes structures and features to detect, assess, impede, and delay threats up to and including the design basis threat for radiological sabotage in compliance with CNSC REGDOC-2.5.2, Section 7.22.1.

### **3.3.8 References**

- 3.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.3-2 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.3-3 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.3-4 USNRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."
- 3.3-5 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.3-6 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.3-7 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.3-8 USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- 3.3-9 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.3-10 10 CFR 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."
- 3.3-11 CSA N289.5, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities," CSA Group.
- 3.3-12 NK054-REP-01210-00098 R000, Geotechnical Data Report – R2, Darlington New Nuclear Project Geotechnical Investigation, EXP Services Inc. Project No. BRM-00025482-A0," Ontario Power Generation. 2013 (Reference 2.7-37)
- 3.3-13 NK054-REP-01210-0418696, "Geologic and Geophysical Evaluation, Darlington Site Investigation – Phase III (Field Work), AMEC Report No. D0053/RP/002 R01, Volumes 1 and 2," Ontario Power Generation. 2012 (Reference 2.7-36)
- 3.3-14 NK054-REP-07730-00005 Rev. R000, Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation. 2009 (Reference 2.7-41)
- 3.3-15 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC. (Reference 2.7-35)

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.3-16 EPRI TR-1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute. 2013 (Reference 2.7-28)
- 3.3-17 Wair, B.R., DeJong, J.T., and Shantz, T., "Guidelines for Estimation of Shear Wave Velocity Profiles," Pacific Earthquake Engineering Research Center, PEER Report 2012/08. (Reference 2.7-47).
- 3.3-18 Campbell, K.W., et. al, Reference-Rock Site Conditions for Central and Eastern North America: Part II – Attenuation (Kappa) Definition, Pacific Engineering Research Center, PEER Report No. 2014/12. 2014 (Reference 2.7-42).
- 3.3-19 EPRI TR-102293-V5, "Guidelines for Determining Design Basis Ground Motions," Electric Power Research Institute. 1993 (Reference 2.7-44)
- 3.3-20 Silva, W.J., N. Abrahamson, G. Toro and C. Costantino. "Description and validation of the stochastic ground motion model." Brookhaven National Laboratory, Associated Universities, Inc. Upton, New York,. 1996 (Reference 2.7-45)
- 3.3-21 CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.3-22 Toro G. R., "Probabilistic models of site velocity profiles for generic and site-specific ground motion amplification studies." Technical Report 779574, Brookhaven National Laboratory, Upton, New York.
- 3.3-23 Darendeli, M.B., "Development of a new family of normalized modulus reduction and material damping curves." PhD thesis, The University of Texas, Austin.
- 3.3-24 ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers.
- 3.3-25 USNRC NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines." Reference 2.7-24
- 3.3-26 NK38-REP-03611-10041 (Reference 2.7-10)
- 3.3-27 USNRC DC/COL-ISG-017, "Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses."
- 3.3-28 NK38-CORR-03611-0847339, "Disposition to the CNSC's Comments on the Submission of an Update to the Probabilistic Seismic Hazard Assessment for Darlington NGS," August 19, 2020.)
- 3.3-29 Lilhanand, K., and Tseng, W.S., "Development and Application of Realistic Earthquake Time Histories Compatible with Multiple-damping Response Spectra," Proceedings of the 9th World Conference on Earthquake Engineering, Tokyo-Kyoto, Japan, Vol. II, 1988.
- 3.3-30 Abrahamson, N., "Non-stationary Spectral Matching, Seismological Research Letters, Vol. 63, No. 1, 1992."
- 3.3-31 Al Atik, L. and Abrahamson, N., "An Improved Method for Nonstationary Spectral Matching, Earthquake Spectra, Vol. 26, No. 3, August 2010."
- 3.3-32 USNRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.3-33 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,"
- 3.3-34 USNRC NUREG/CR-7193, "Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors (SMRs)."
- 3.3-35 ACI 350.3-20, "Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures (ACI 350.3-20) and Commentary," American Concrete Institute.
- 3.3-36 ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," American Institute of Steel Construction.
- 3.3-37 NEDC-33926P, "Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.3-38 BC-TOP-4A, "Seismic Analyses of Structures and Equipment for Nuclear Power Plants," Bechtel Power Corporation.
- 3.3-39 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.3-40 USNRC Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes."
- 3.3-41 USNRC Regulatory Guide 1.166, "Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake."
- 3.3-42 ANSI/ANS-2.23, "Nuclear Power Plant Response to an Earthquake," American National Standards Institute, Inc./American Nuclear Society.
- 3.3-43 USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants."
- 3.3-44 ASCE/SEI 7, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," American Society of Civil Engineers.
- 3.3-45 USNRC DC/COL-ISG-7, "Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures."
- 3.3-46 BC-TOP-3-A, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Bechtel Power Corporation.
- 3.3-47 USNRC Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants."
- 3.3-48 ORNL-NSIC-5, "U.S. Reactor Containment Technology. A Compilation of Current Practice in Analysis, Design, Construction, Test, and Operation," Oak Ridge National Laboratory.

**Table 3.3-1: Seismic Categories and Design Basis of BWRX-300 Structures**

Structure	Safety Class	Seismic Category /Evaluation	Design/Evaluation Basis	Design Basis Earthquake <sup>(1)</sup>	Limit State <sup>(2)</sup>
SCCV and Containment Steel Structures	SC1	Seismic Category A	CSA N289 series ASCE/SEI 43 and ASCE/SEI 4 ASME BPVC (see NEDC-33926P)	DBE	LS-D
Containment Internal Structures	SC1	Seismic Category A	CSA N289 series and N291 ASCE/SEI 43 and ASCE/SEI 4 ANSI/AISC N690	DBE	LS-D
RB SC and Steel Structures	SC1	Seismic Category A			
RWB Structure	SC3 <sup>(3)</sup>	Seismic Category RW-IIa	CSA N289 Series and N291 RG 1.143 ASCE/SEI 43 and ASCE/SEI 4	½ DBE	LS-D
		Seismic Interaction Evaluation		DBE	LS-C
CB Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
TB Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 and CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Reactor Auxiliary Bay Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 and CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Other Structures	SC3/S CN	Non-Seismic Category	NBC		

1. DBE is defined in Subsection 3.3.1

2. Limit States per ASCE/SEI 43:

- LS-D Essentially elastic response
- LS-C response with limited permanent deformations

3. The RWB is designed in accordance with the radioactive waste management requirements for Category RW-IIa from U.S. NRC RG 1.143

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-2: Base Case Rock Dynamic Properties**

Bedrock Formation	Total Unit Weight (kN/m <sup>3</sup> )	Base Case Shear Wave			Poisson's Ratio
		V <sub>s</sub> (m/s)	$\sigma_{\mu \ln}$	$\sigma_{\mu \ln V_s}$	
Blue Mountain (Whitby)	26.4	2,203	0.10	0.15	0.30
Lindsay1	26.6	2,708	0.10	0.15	0.31
Lindsay2	26.6	2,591	0.10	0.15	0.31
Lindsay3	26.6	2,881	0.10	0.15	0.31
Verulam1	26.4	2,185	0.10	0.15	0.33
Verulam2	26.4	2,500	0.10	0.15	0.31
Verulam3	26.4	2,623	0.10	0.15	0.31
Verulam4	26.4	2,761	0.10	0.15	0.31
Bobcaygeon	26.3	2,906	0.10	0.15	0.31
Gull River	26.5	3,139	0.10	0.15	0.32
Shadow Lake	25.7	2,706	0.10	0.15	0.30
Gneiss	27.3	3,128	0.10	0.15	0.28

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-3: Base Case Engineered Fill and In-situ Soil Dynamic**

Layer	Shear Wave Velocity (m/s)			Poisson's Ratio
	Base Case $V_s$	$\sigma_\mu$ In	$\sigma_\mu$ In $V_s$	Average
Fill 1	207	0.40	0.25	0.35
Fill 2	235	0.40	0.25	0.35
Fill 3	254	0.40	0.25	0.35
Fill 4	271	0.40	0.25	0.35/0.40
Fill 5	287	0.40	0.25	0.35/0.40
Fill 6	300	0.40	0.25	0.35/0.40
Fill 7	314	0.40	0.25	0.35/0.40
Upper till	513	0.40	0.25	0.35/0.40
Intermediate glacio-lacustrine (Sandy)	506	0.40	0.15	0.40
Intermediate glacio-lacustrine (Silty)	480	0.40	0.15	0.40
Lower till	524	0.40	0.15	0.40

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-4: Rock Layers Kappa Values**

<b>Case</b>	<b>Bedrock Kappa (<math>\kappa_0</math>, ref; sec)</b>	<b>Rock Layer Kappa (<math>\kappa_r</math>; sec)</b>	<b>Total Kappa at Top of Rock (sec)</b>
Base Case	0.006	0.002	0.008
Lower Realization	0.006	0	0.006
Upper Realization	0.006	0.006	0.012



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-5: Selected Time History Records**

Record	NUREG/CR-6728 Database Bin	Component	Scaling Factor	HP (Hz)	LP (Hz)	Peak Ground Acceleration (G)	Duration (seconds)
HWA056	Rock M7+ R 10-50 km	H1 (North)	1.46	0.03	50	0.203	86.000
		H2 (West)	1.46	0.02	50	0.207	86.000
		Vertical	1.50	0.02	50	0.120	86.000
TCU047	Rock M7+ R 10-50 km	H1 (North)	0.40	0.03	50	1.168	89.995
		H2 (West)	0.44	0.02	50	0.700	89.995
		Vertical	0.50	0.02	50	0.556	89.995
ILA063	Rock M7+ 50-100 km	H1 (North)	1.31	0.02	50	0.221	78.990
		H2 (West)	1.37	0.02	50	0.226	78.990
		Vertical	2.26	0.04	50	0.122	78.990
HWA026	Rock M7+ 50-100 km	H1 (North)	2.10	0.03	50	0.135	89.995
		H2 (West)	1.45	0.02	50	0.202	89.995
		Vertical	2.40	0.02	50	0.110	89.995
TAP075	Rock M7+ 100-200 km	H1 (North)	1.69	0.02	50	0.171	91.999
		H2 (West)	1.45	0.01	30	0.205	91.999
		Vertical	2.57	0.03	30	0.110	91.999

**Table 3.3-6: Seismic Damping Values for BWRX-300 Structures**

<b>Material</b>	<b>Response Level 1</b>	<b>Response Level 2</b>
Steel-plate composite structures	3	5
Welded and Friction-bolted steel structures	2	4
Bearing-bolted steel structures	4	7
Reinforced concrete structures	4	7

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-7: Seismic Damping Values for Primary Subsystems**

Component	Level 1 Damping		Level 2 Damping	
	Horizontal	Vertical	Horizontal	Vertical
Reactor Vessel	2	2	4.0	4.0
Vessel Support Skirt	2	2	4.0	4.0
Shroud	2	2	4.0	4.0
Shroud Support Spring	2	2		
Shroud Head & Separator	2	2	4.0	4.0
Fuel	4	4	6.0	6.0
CRD Guide Tubes	1	1	2.0	2.0
CRD Housing	1	1	2.0	2.0
CRD Restraint Springs	-	2		
Stabilizer and Bellows	-	2		
Welded Steel			4.0	4.0
Bolted Steel			7.0	7.0

**Table 3.3-8: Preliminary Dynamic Loading  $\alpha$ ,  $\beta$  – Damping**

Loading	Shell Model	Total Model Damping at A & B Freq	A Freq (Hz)	B Freq (Hz)	$\alpha$	$\beta$
LOCA	52	6%	10	60	6.527	.000257
		6%	1.8	12.7	1.2083	.0011637
	110	4%	10	60	4.3731	.0001655
		4%	1.8	12.7	.8121	.0007246

**Table 3.3-9: Seismic Damping Values for Piping and Equipment**

<b>Structure or Component</b>	<b>Level 1 Damping</b>	<b>Level 2 Damping</b>
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	3	5
Small-diameter piping systems, diameter equal to or less than 12 in.	2	5
Welded steel structures	3	4
Bolted steel structures	4	7

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.3-10: Piping and Equipment Damping Values for All Other Non-Seismic Loadings**

<b>Structure or Component</b>	<b>When considered by itself and/or combined with other load and designated as normal, upset and emergency</b>	<b>When considered by itself and/or combined with other load and designated as faulted</b>
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	2	3
Small-diameter piping systems, diameter equal to or less than 12 in.	1	2
Welded steel structures	2	4
Bolted steel structures	4	7

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

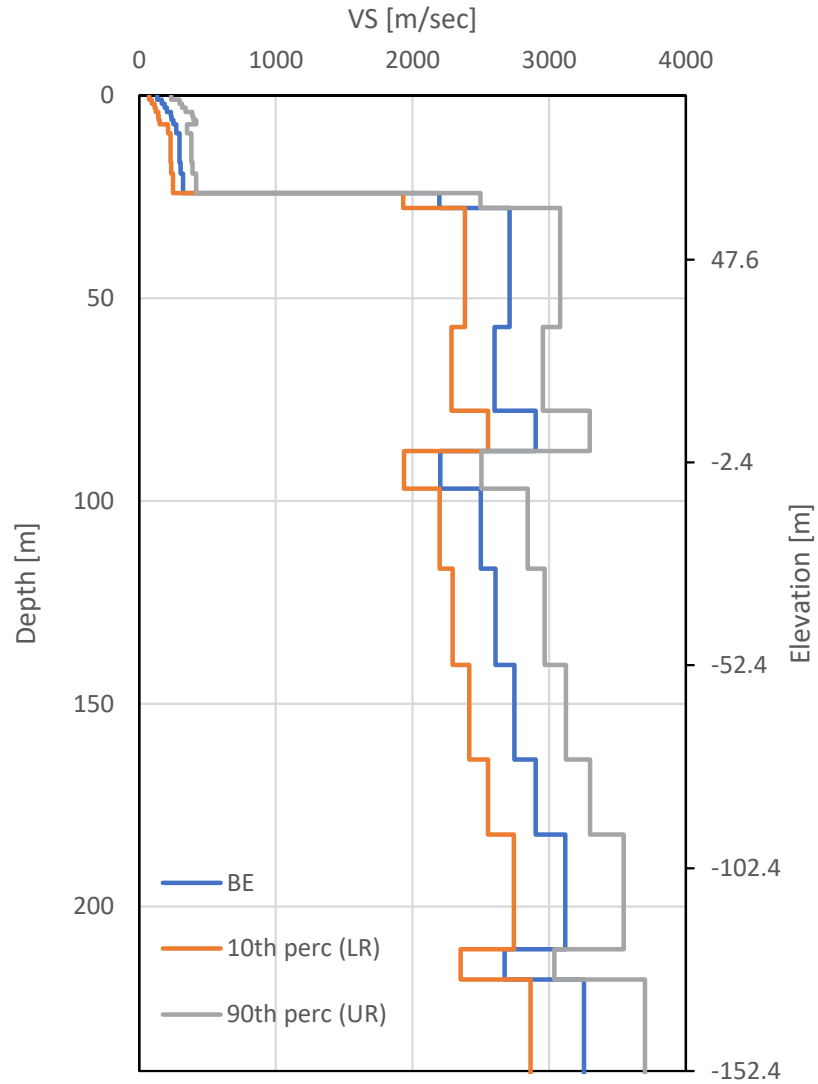


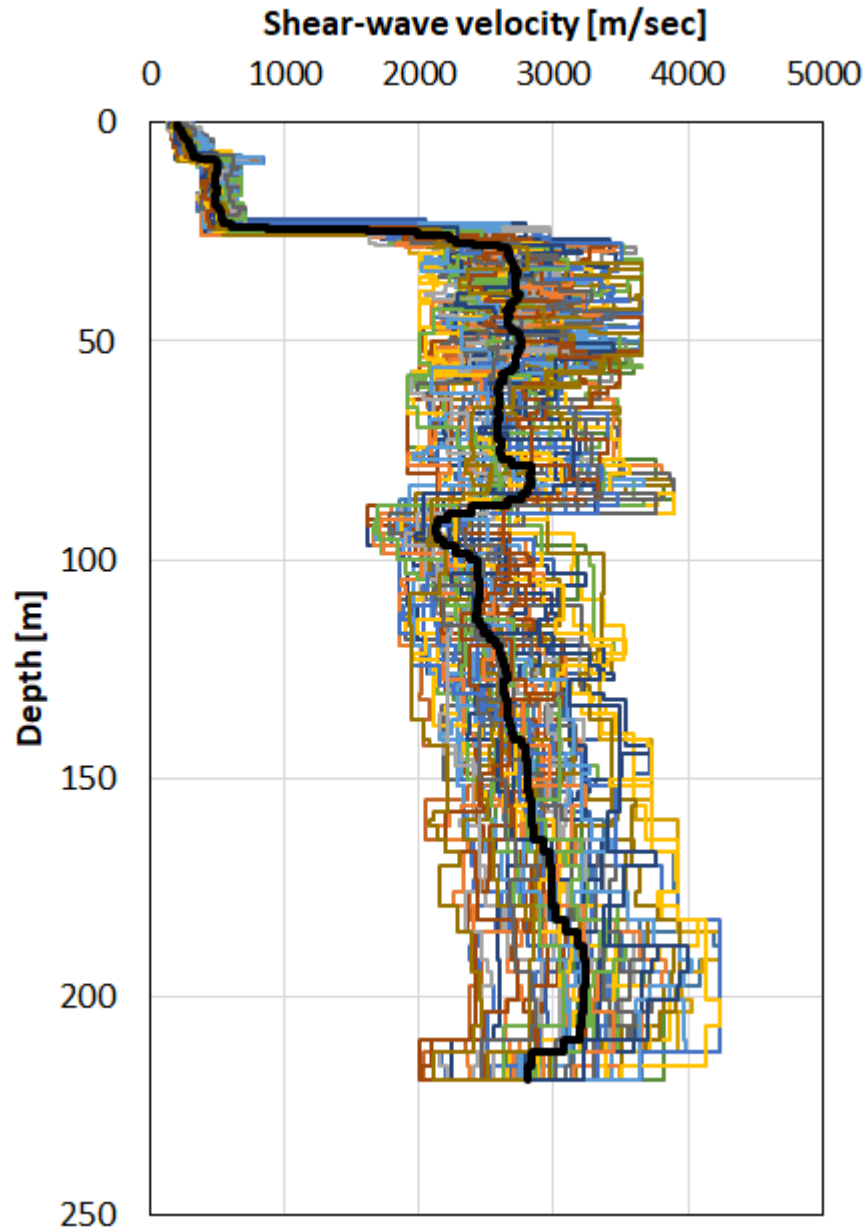
Figure 3.3-1: Shear Wave Velocities for the Bounding In-situ Profile

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

<u>Shear-wave velocity cases (weight)</u>	<u>Kappa cases (weight)</u>	<u>Total weight</u>
BE (0.4)	BE (0.4)	0.16
	10 <sup>th</sup> perc. (0.3)	0.12
	90 <sup>th</sup> perc. (0.3)	0.12
10 <sup>th</sup> perc. (0.3)	BE (0.4)	0.12
	10 <sup>th</sup> perc. (0.3)	0.09
	90 <sup>th</sup> perc. (0.3)	0.09
90 <sup>th</sup> perc. (0.3)	BE (0.4)	0.12
	10 <sup>th</sup> perc. (0.3)	0.09
	90 <sup>th</sup> perc. (0.3)	0.09
		$\Sigma = 1.0$

**Figure 3.3-2: Cases Considered for Explicit Considerations of Epistemic Uncertainties**

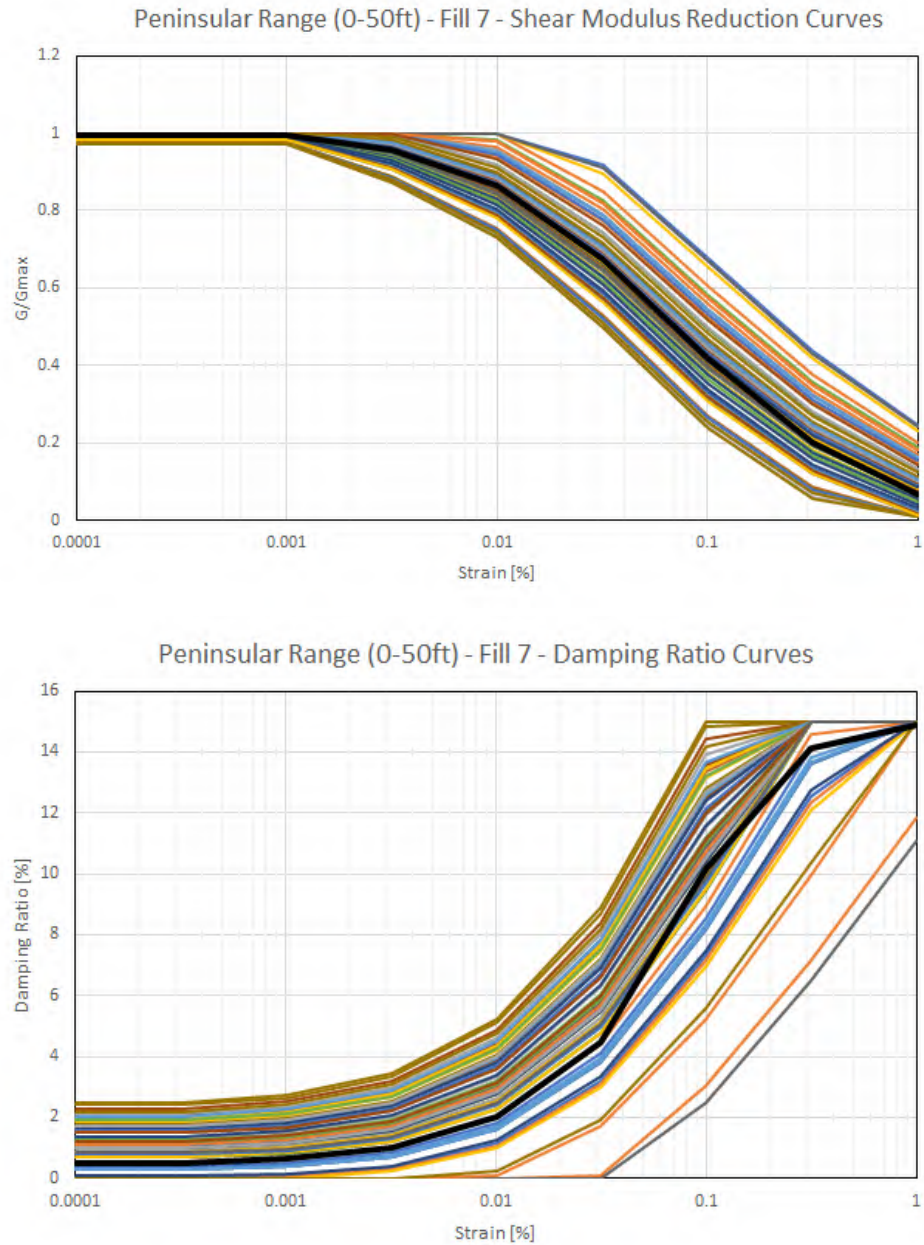




**Figure 3.3-3: Shear Wave Velocity and Layer Thicknesses Randomization – BE-BE Case**

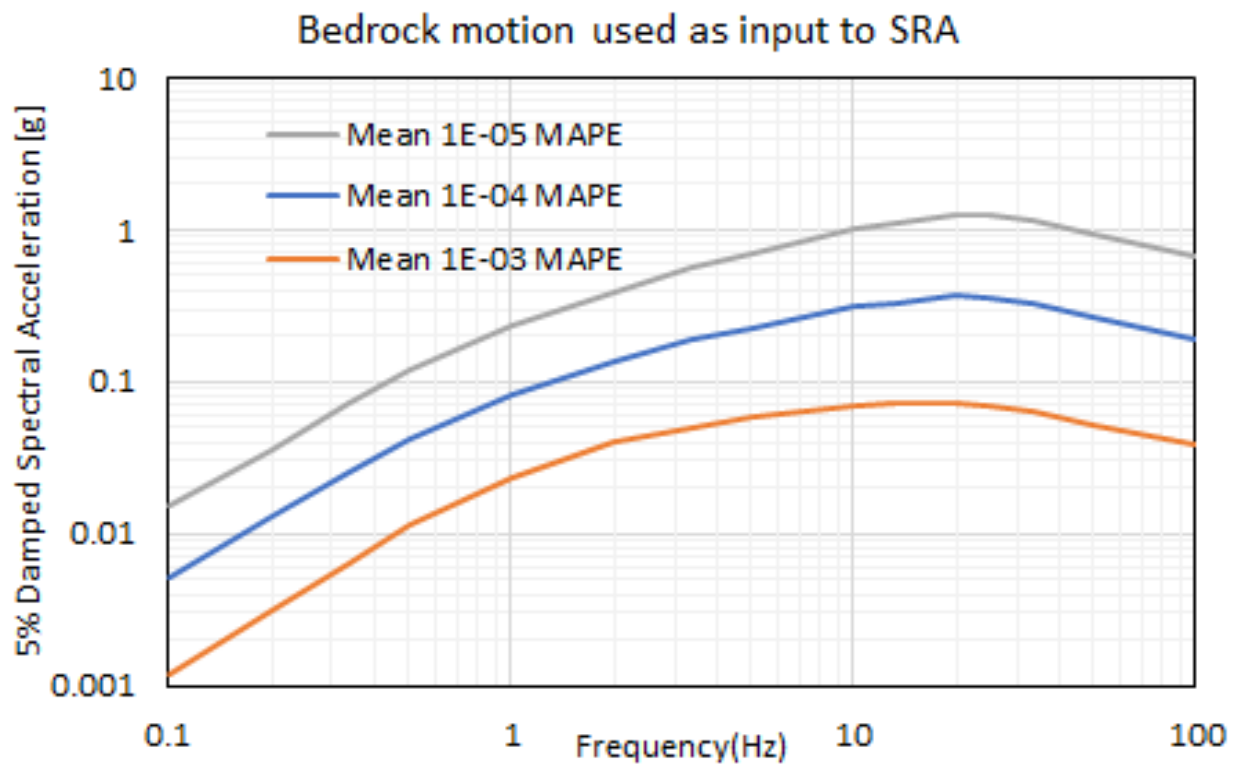
Note: The Black line designates the resulting mean curve

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-4: Soil Degradation Curves Randomization**

Note: The Black line designates the resulting mean curve



**Figure 3.3-5: Uniform Hazard Response Spectra at Bedrock Elevation**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

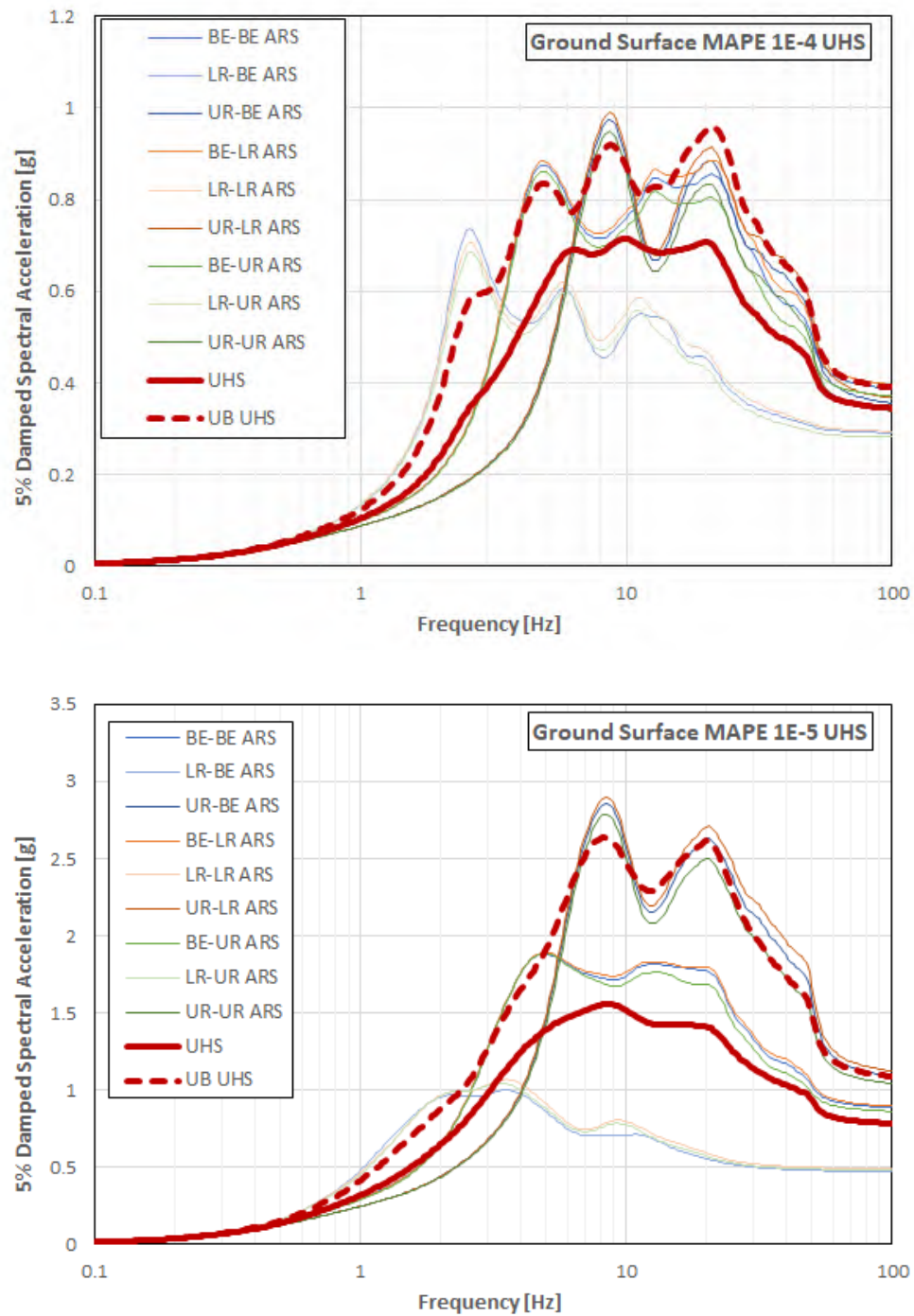


Figure 3.3-6: Ground Surface Uniform Hazard Response Spectra

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

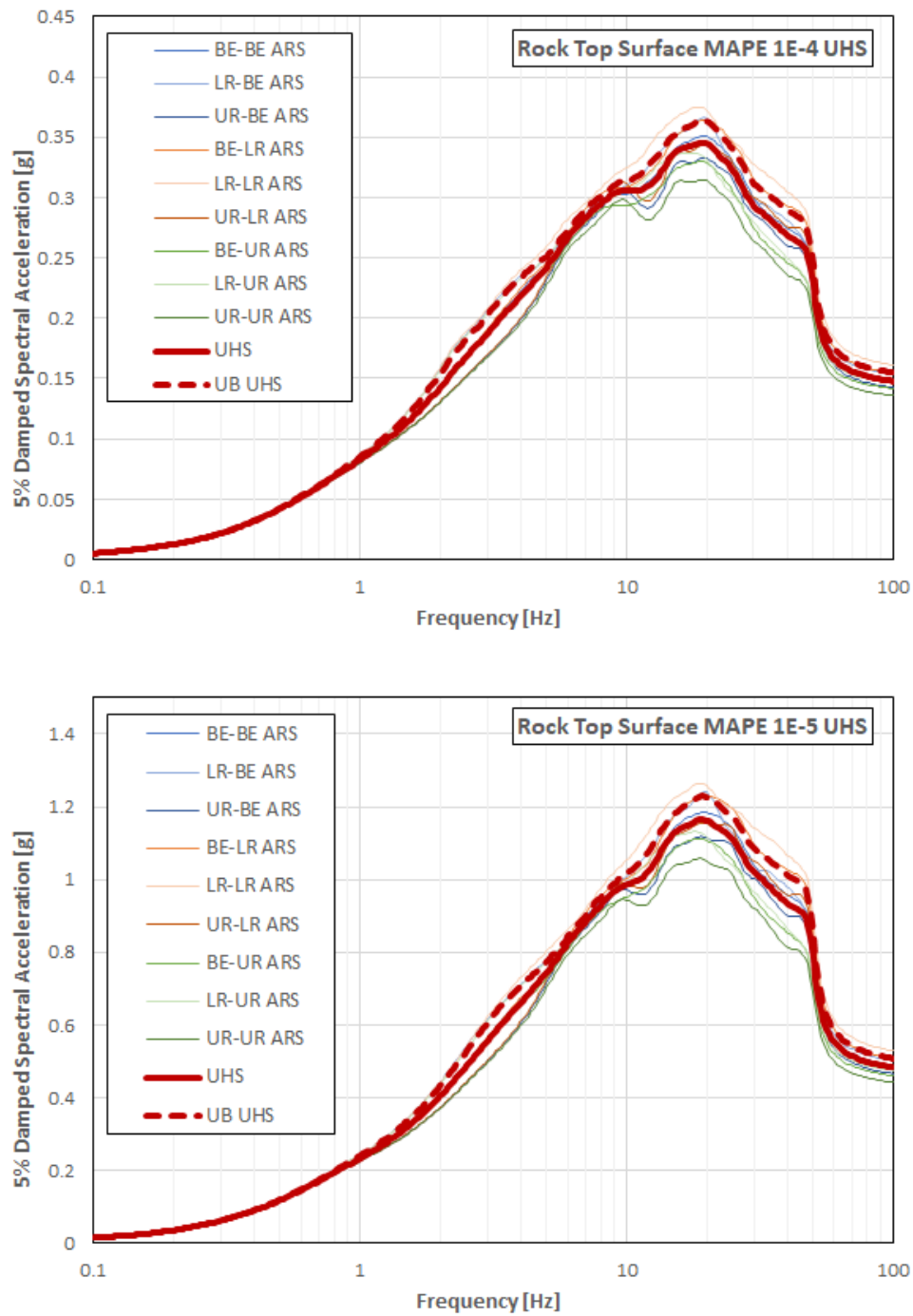
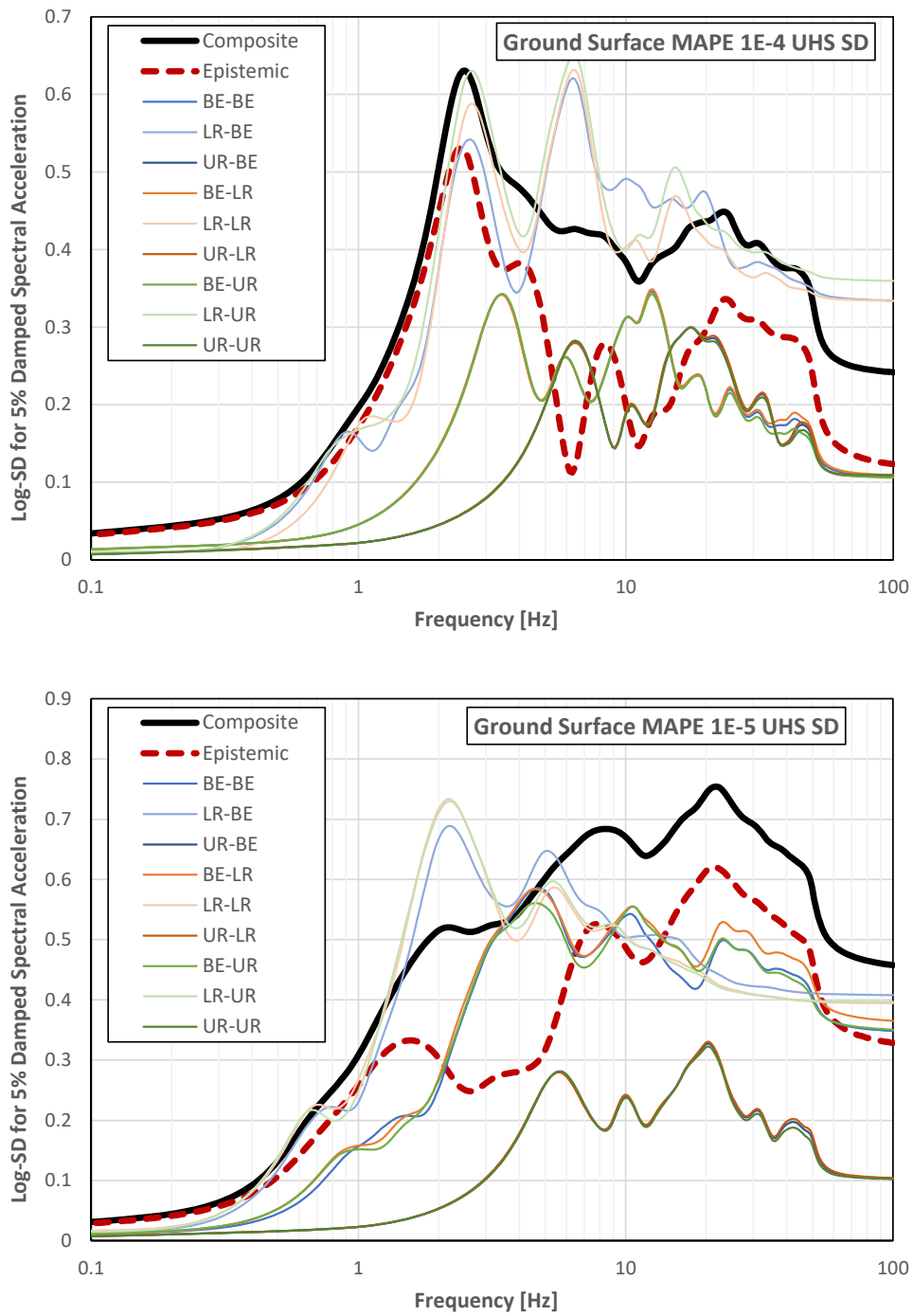


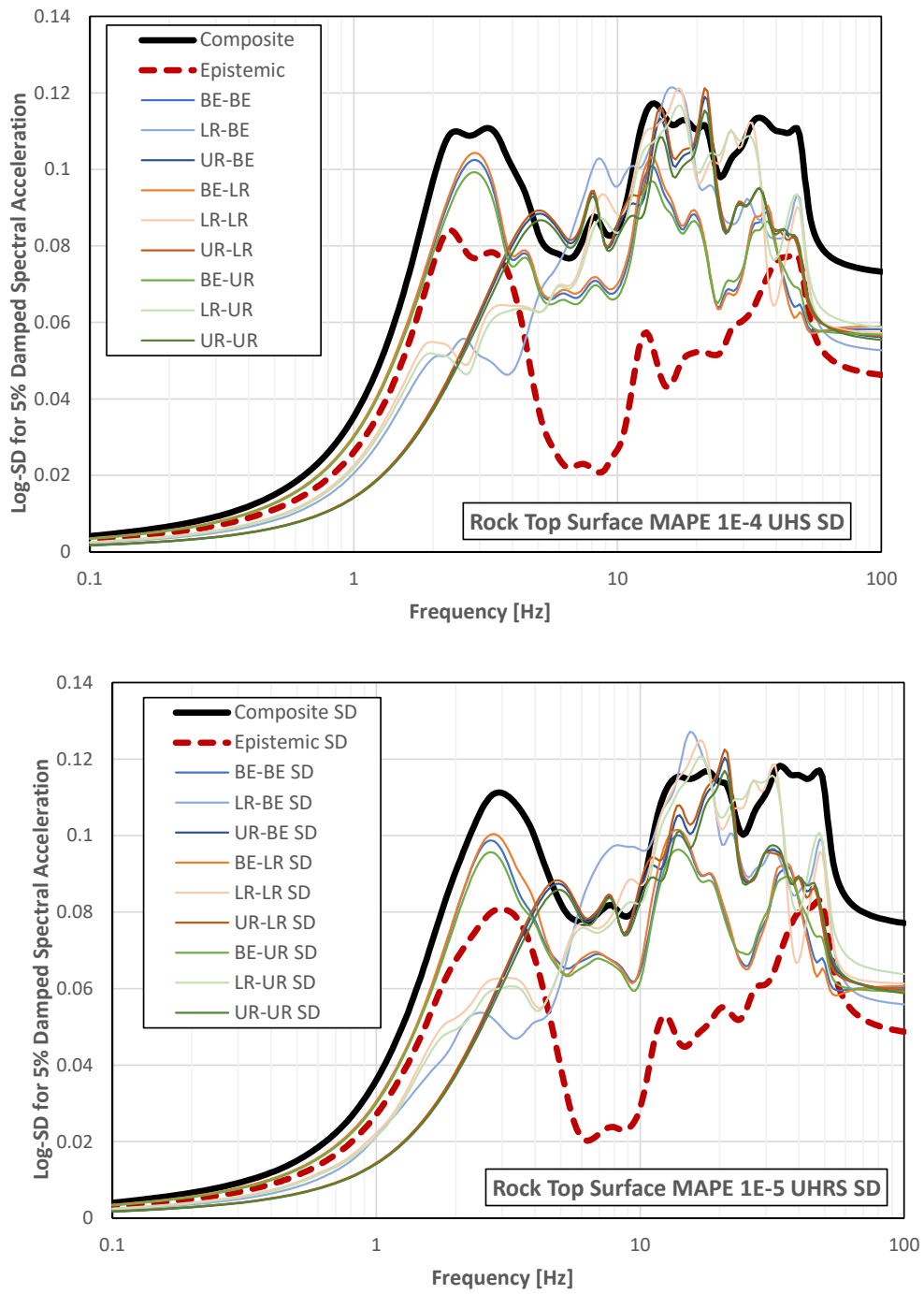
Figure 3.3-7: Rock Top Surface Uniform Hazard Response Spectra

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-8: Ground Surface Composite and Epistemic Log-Normal Standard Deviations**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-9: Rock Top Surface Composite and Epistemic Log-Normal Standard Deviations**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

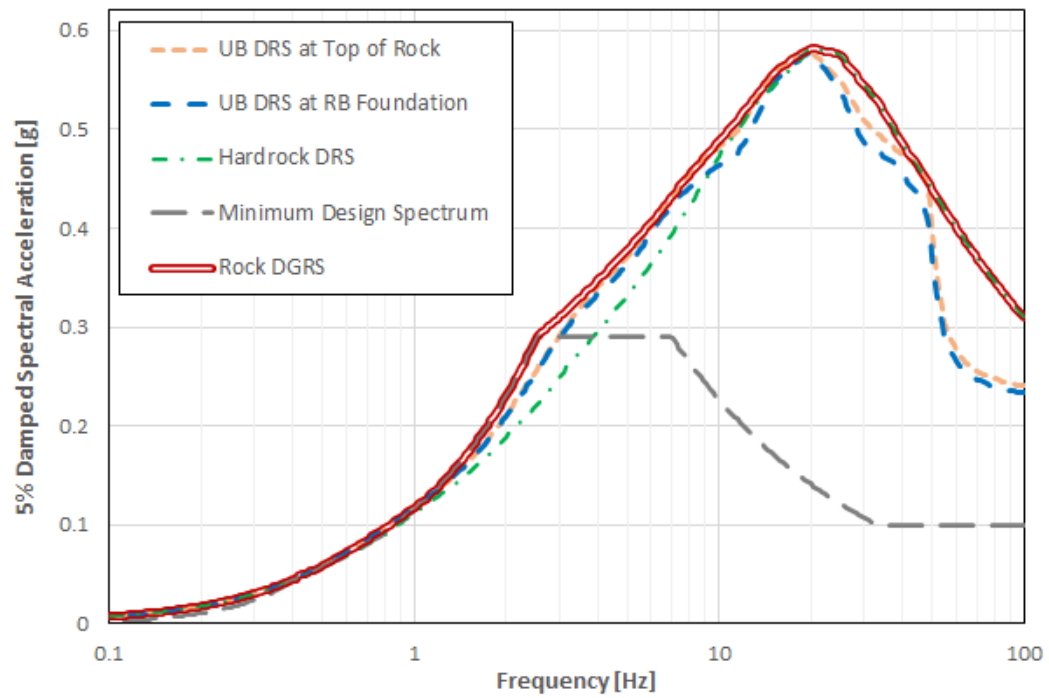
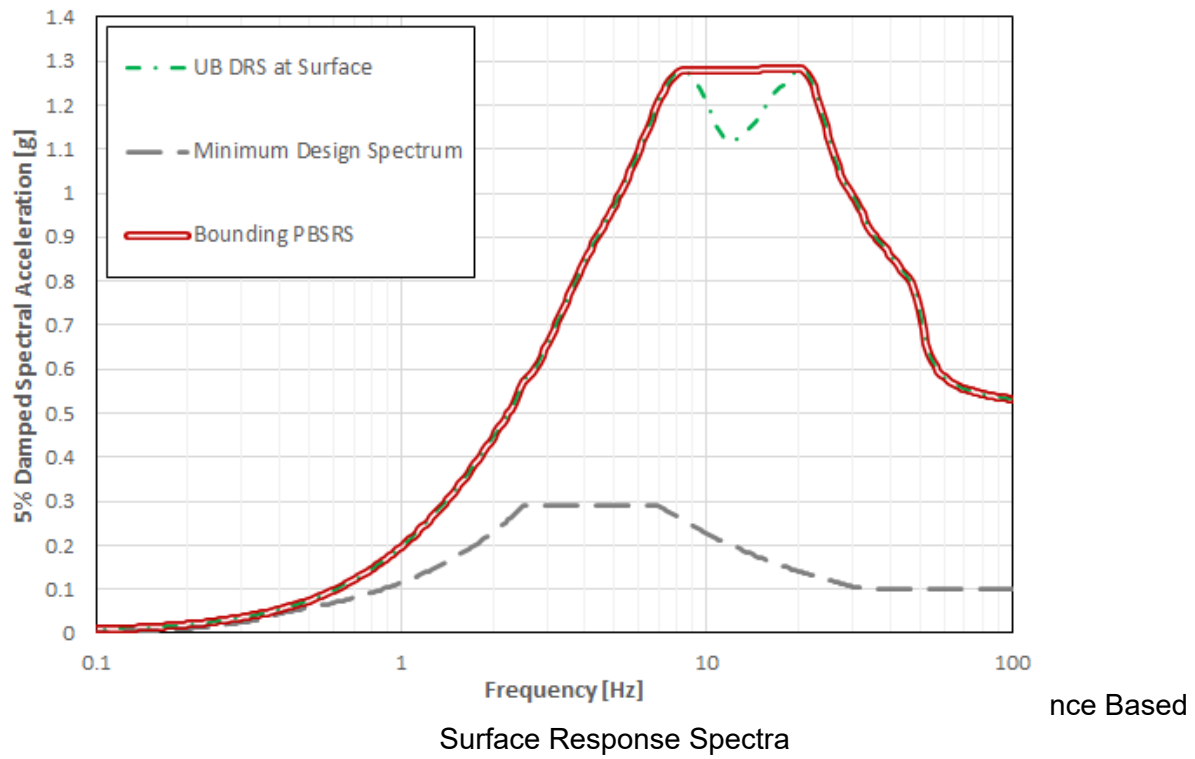
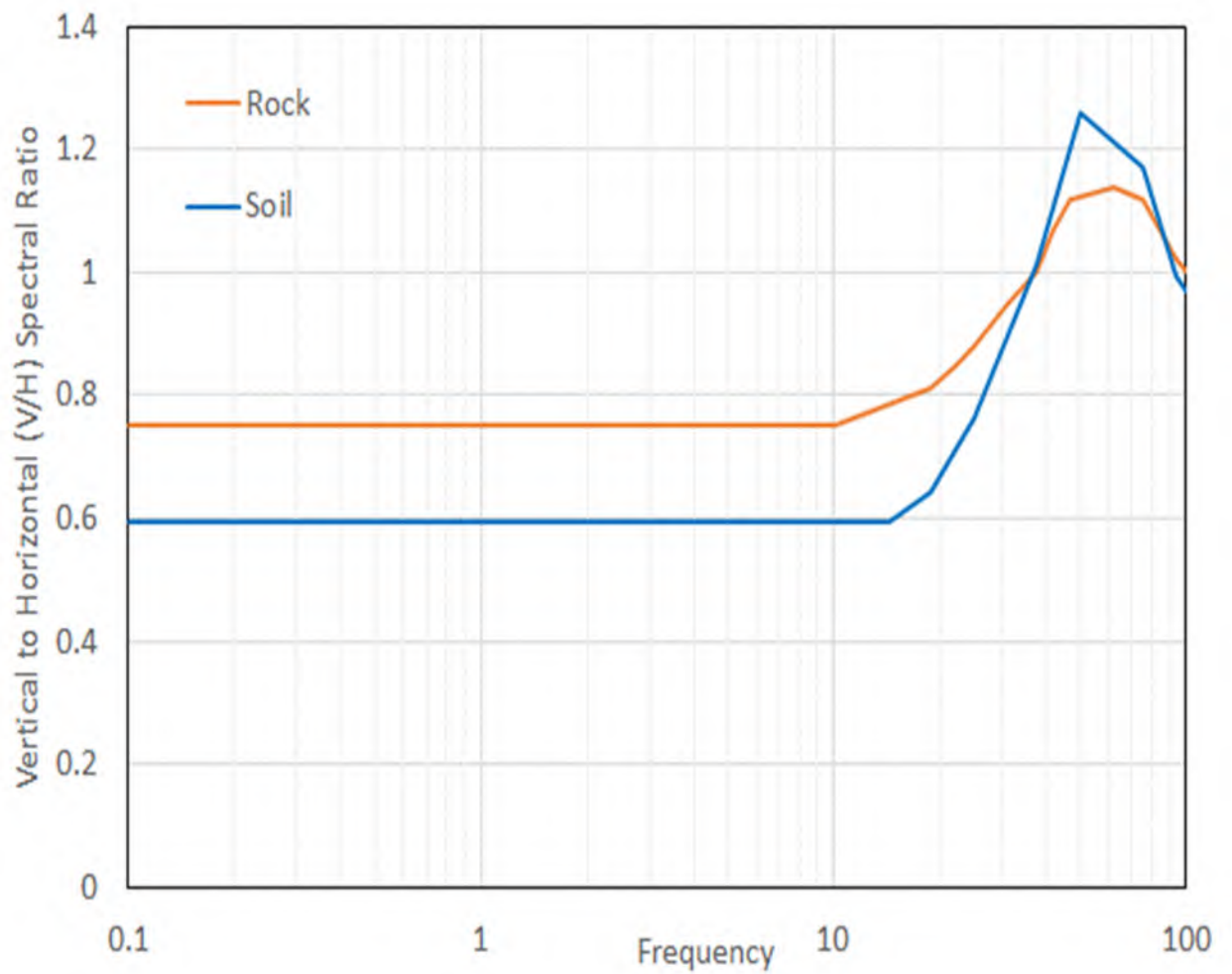


Figure 3.3-10: Bounding Horizontal Rock Ground Motion Response Spectra





**Figure 3.3-11: Bounding Horizontal Performance Based Surface Response Spectra**



**Figure 3.3-12: Vertical to Horizontal (V/H) Spectral Ratios**

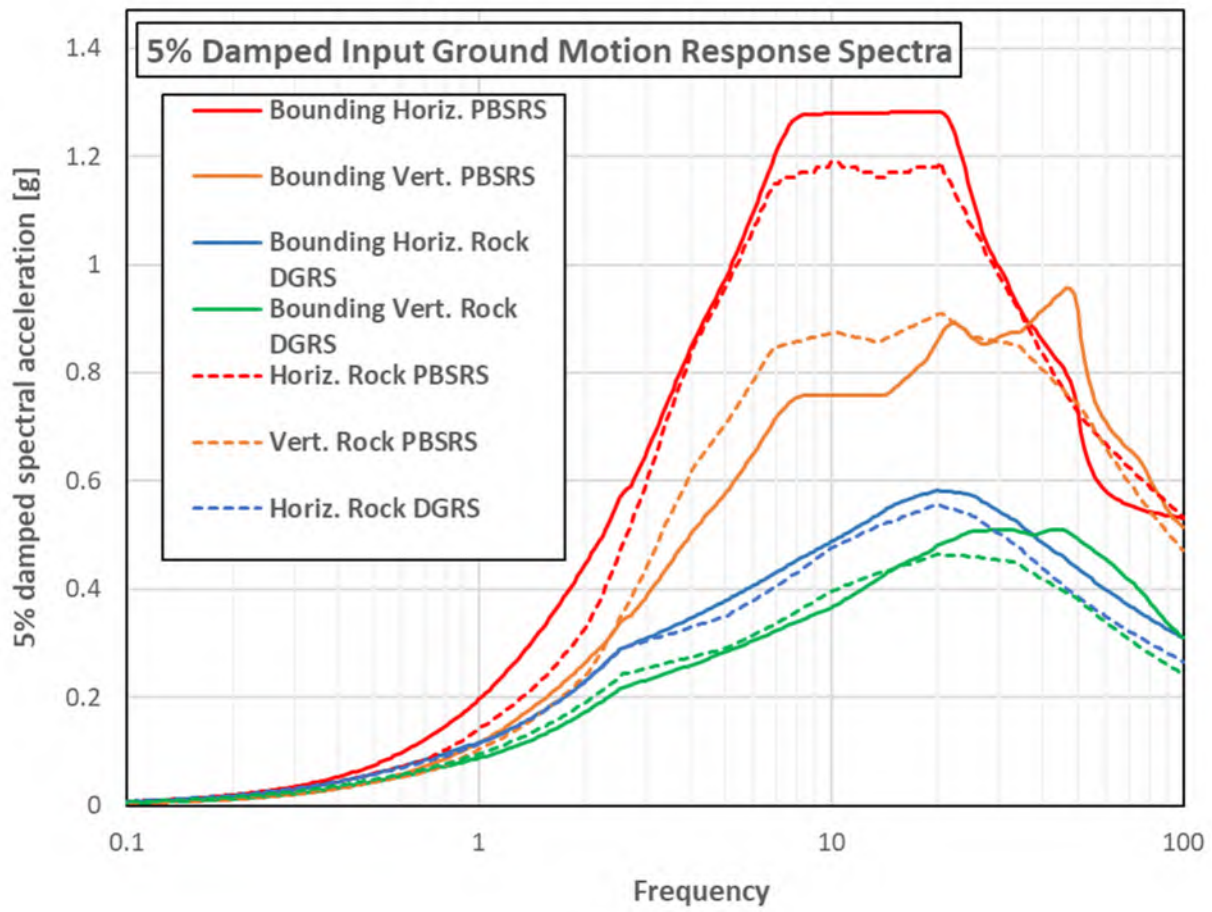
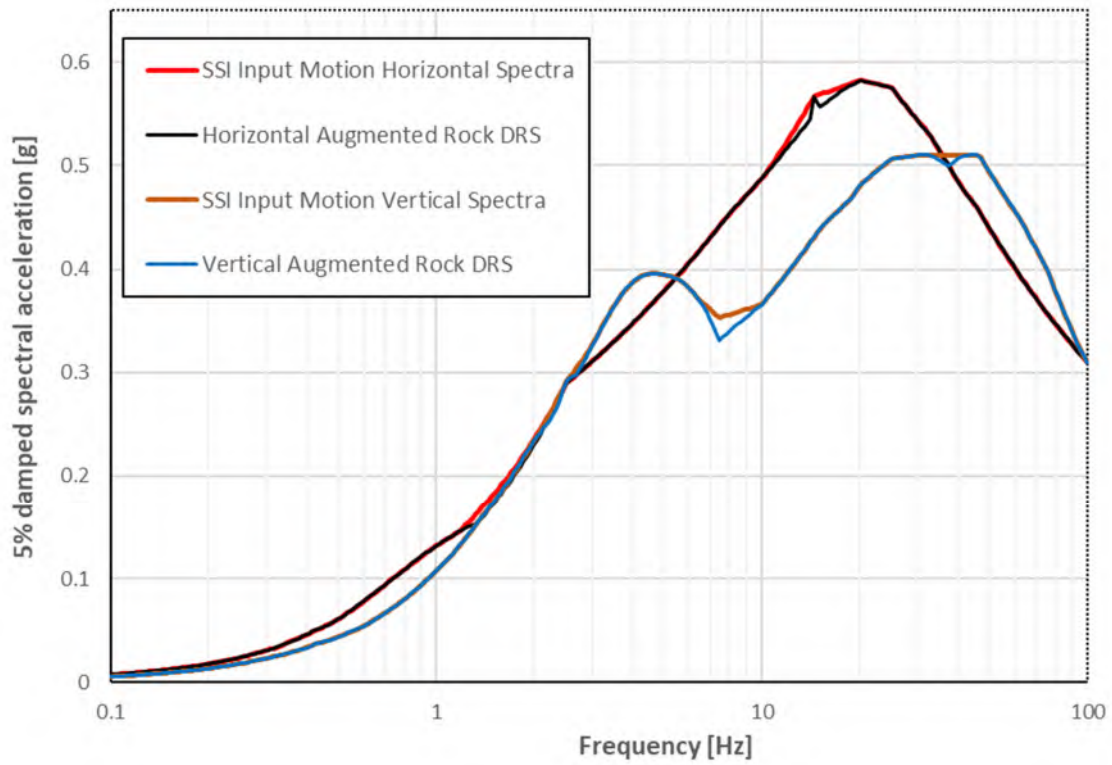
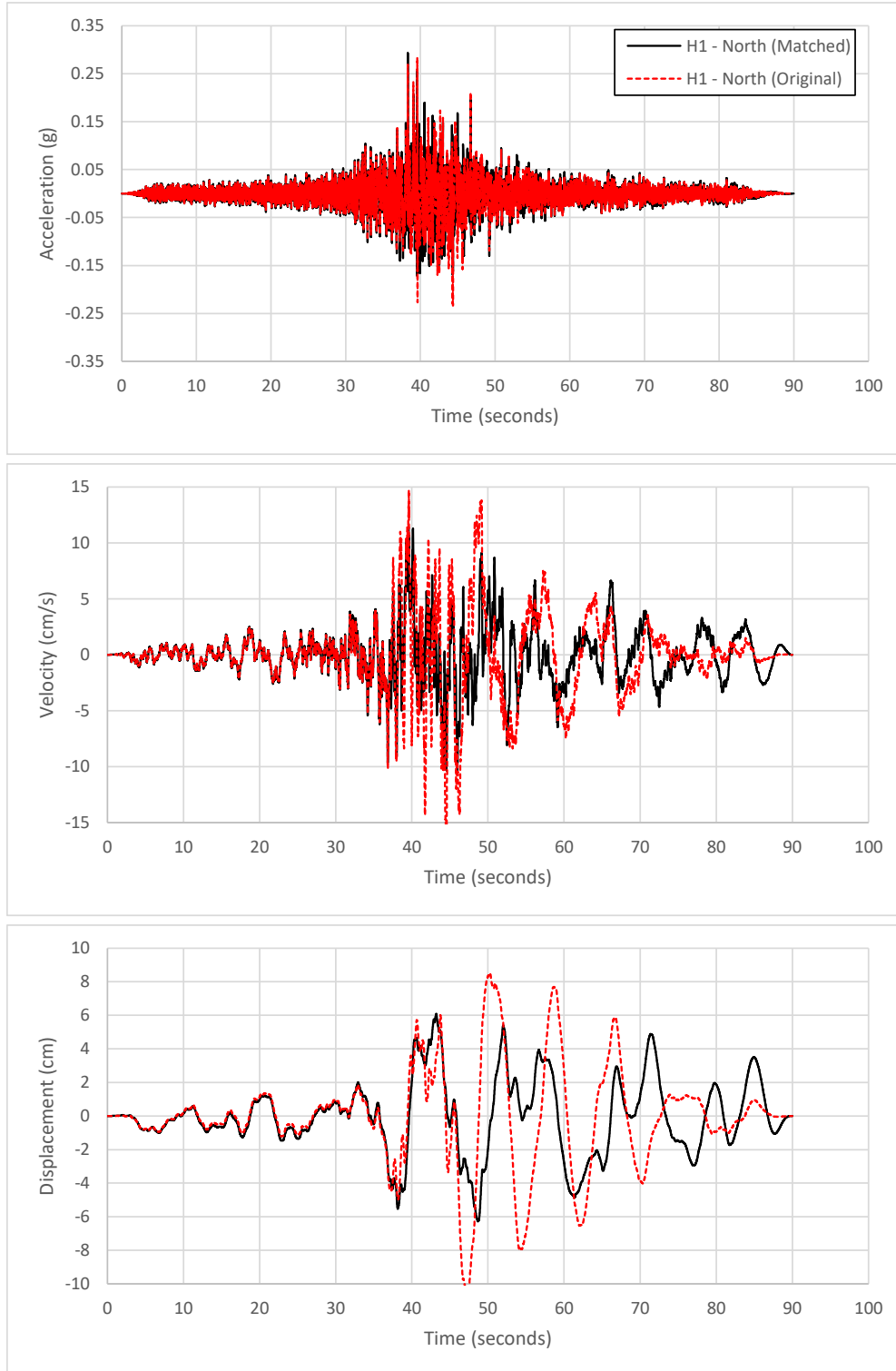


Figure 3.3-13: Comparison of Bounding to Updated Ground Motion Design Response Spectra



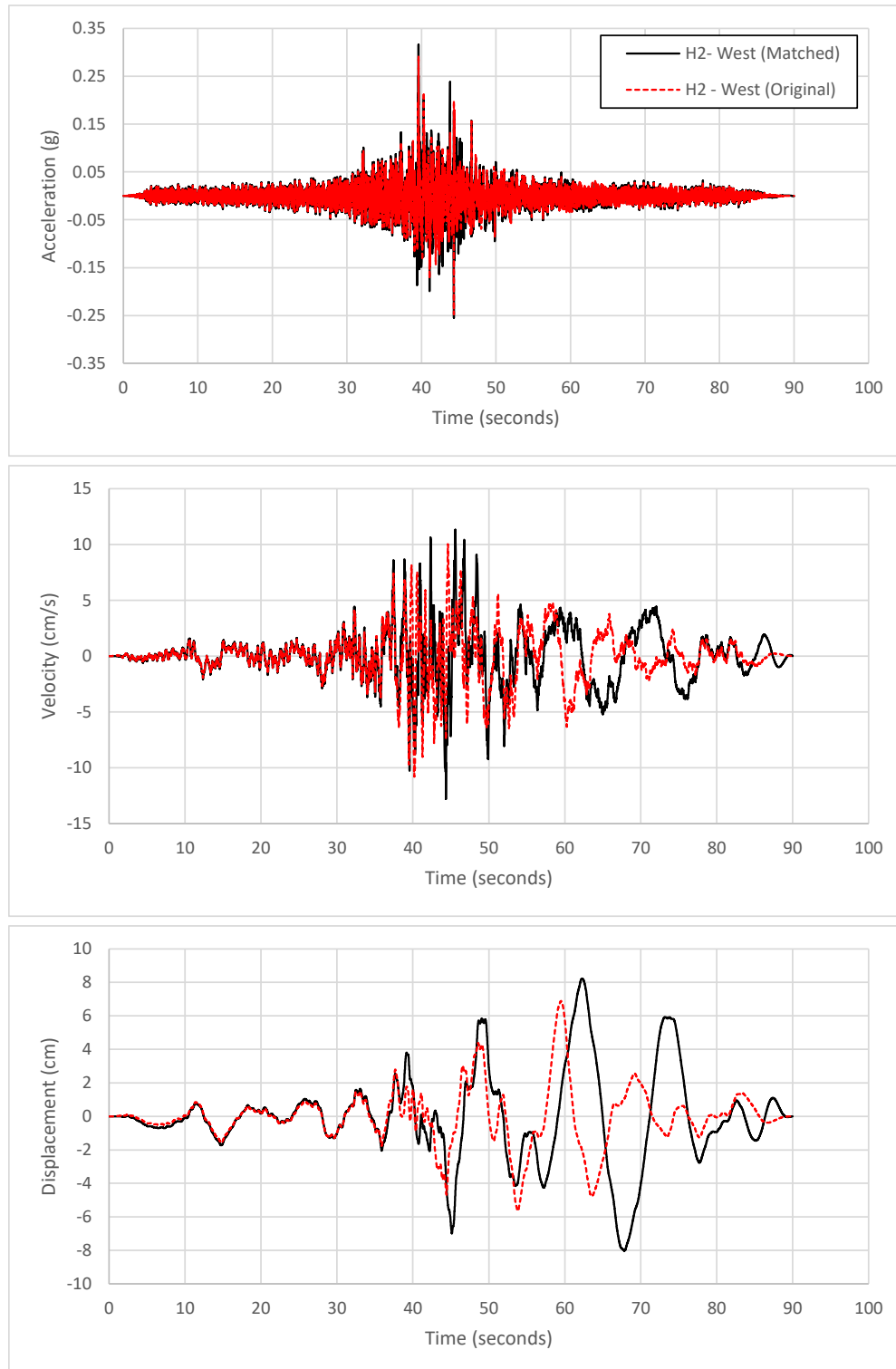
**Figure 3.3-14: Augmented and Smoothed Horizontal and Vertical Rock Design Ground Response Spectra**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



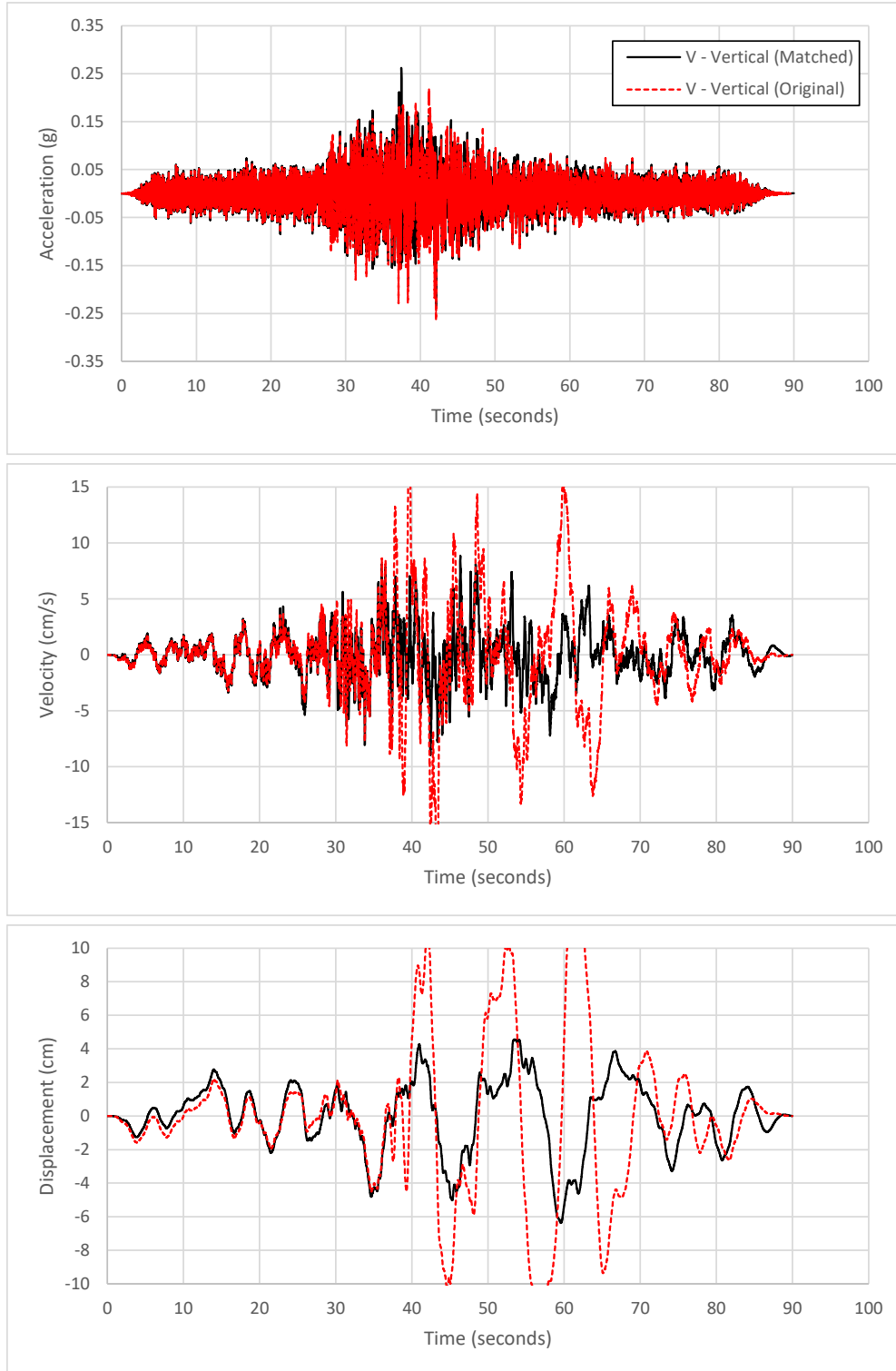
**Figure 3.3-15: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 North**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



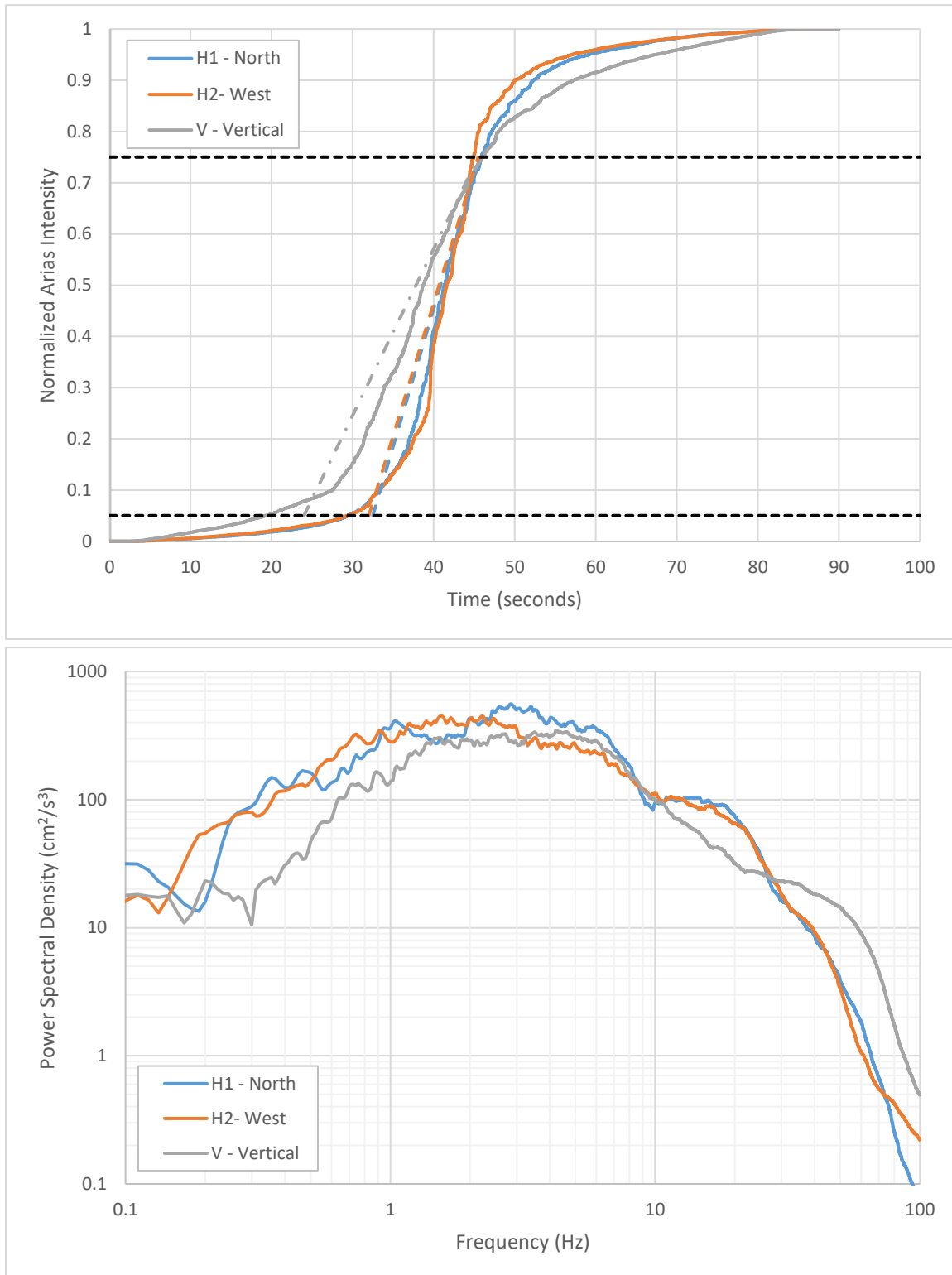
**Figure 3.3-16: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 West**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-17: Vertical Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 Vertical**

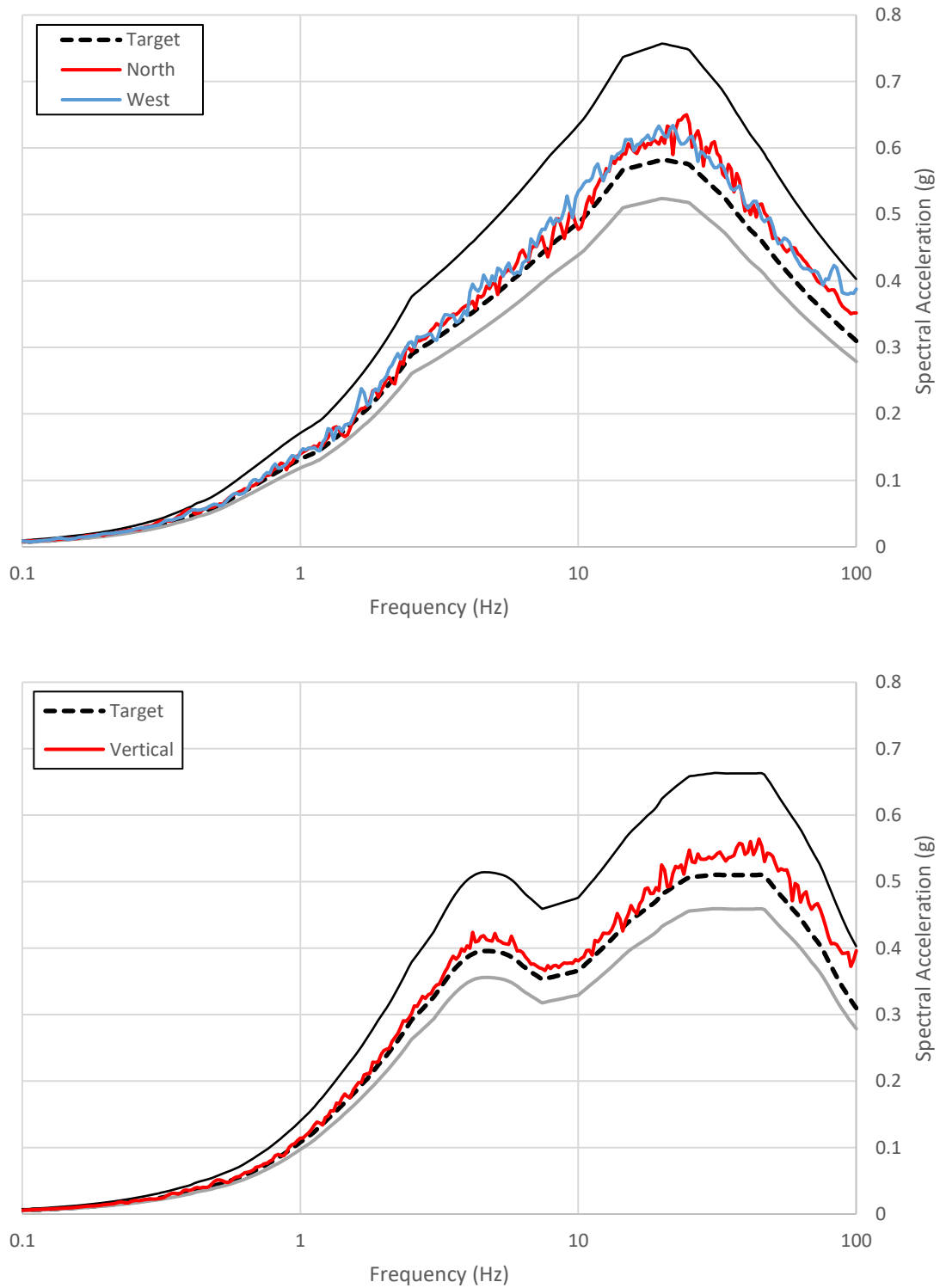
NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-18: Normalized Arias Intensity and Power Spectral Density Function for Response Spectrum Matched HWA026 Acceleration Time Histories**

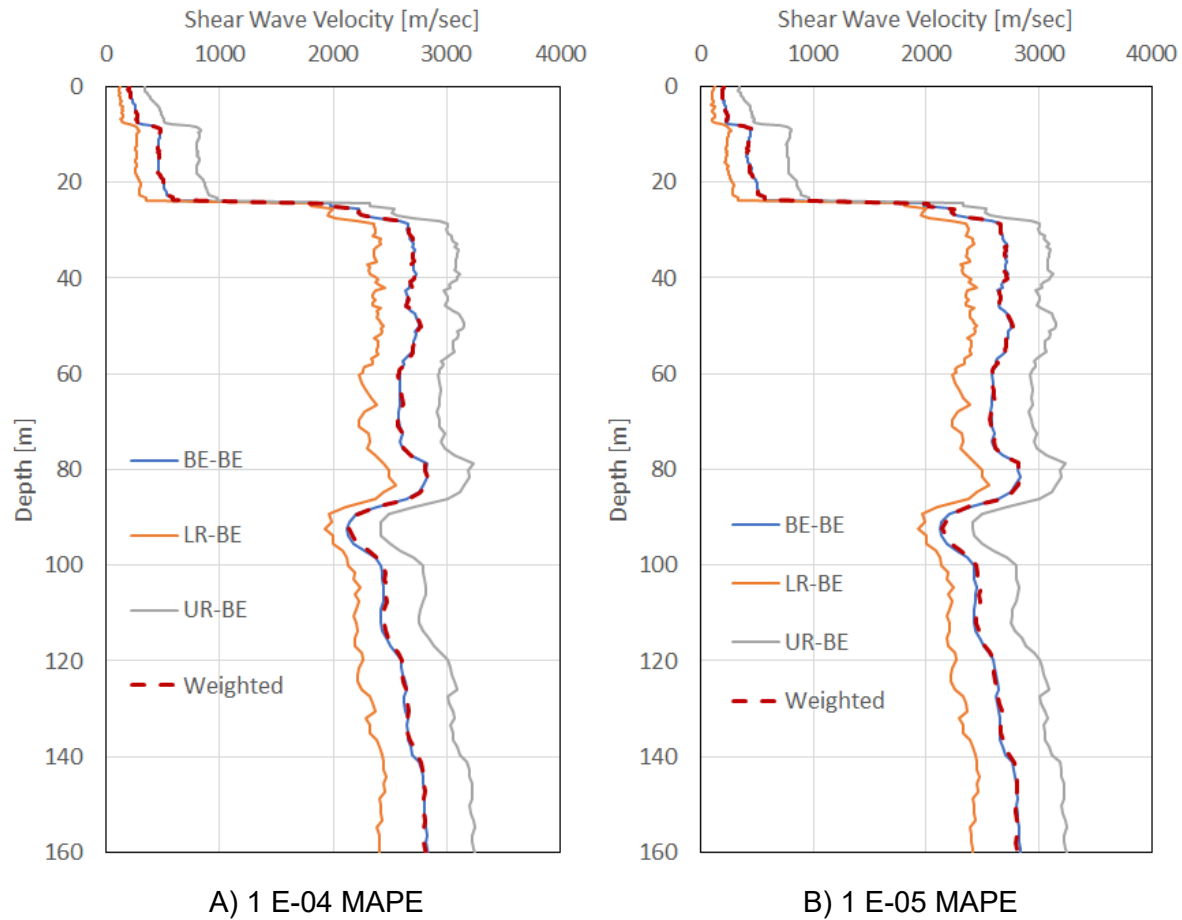


NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

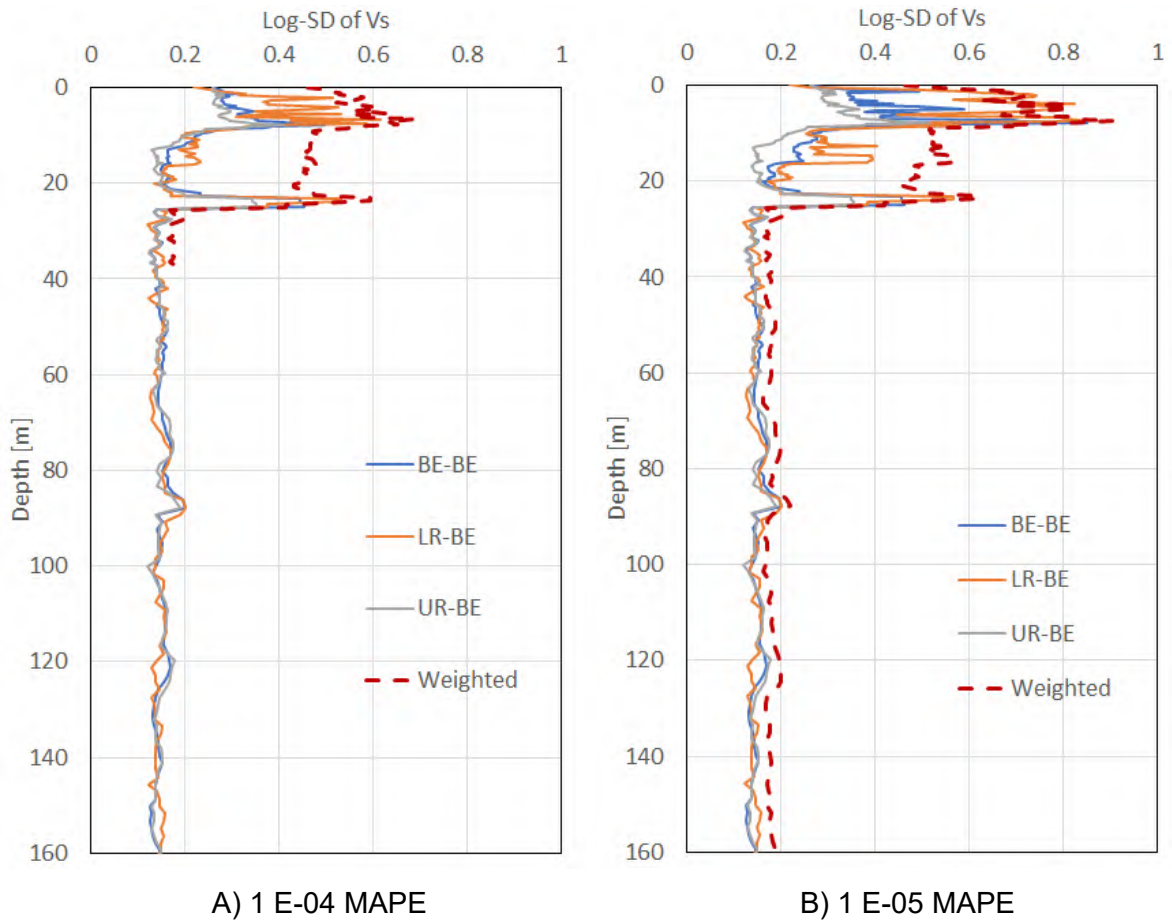


**Figure 3.3-19: 5% Damped Response Spectra for Response Spectrum Matched HWA026 Acceleration Time Histories**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

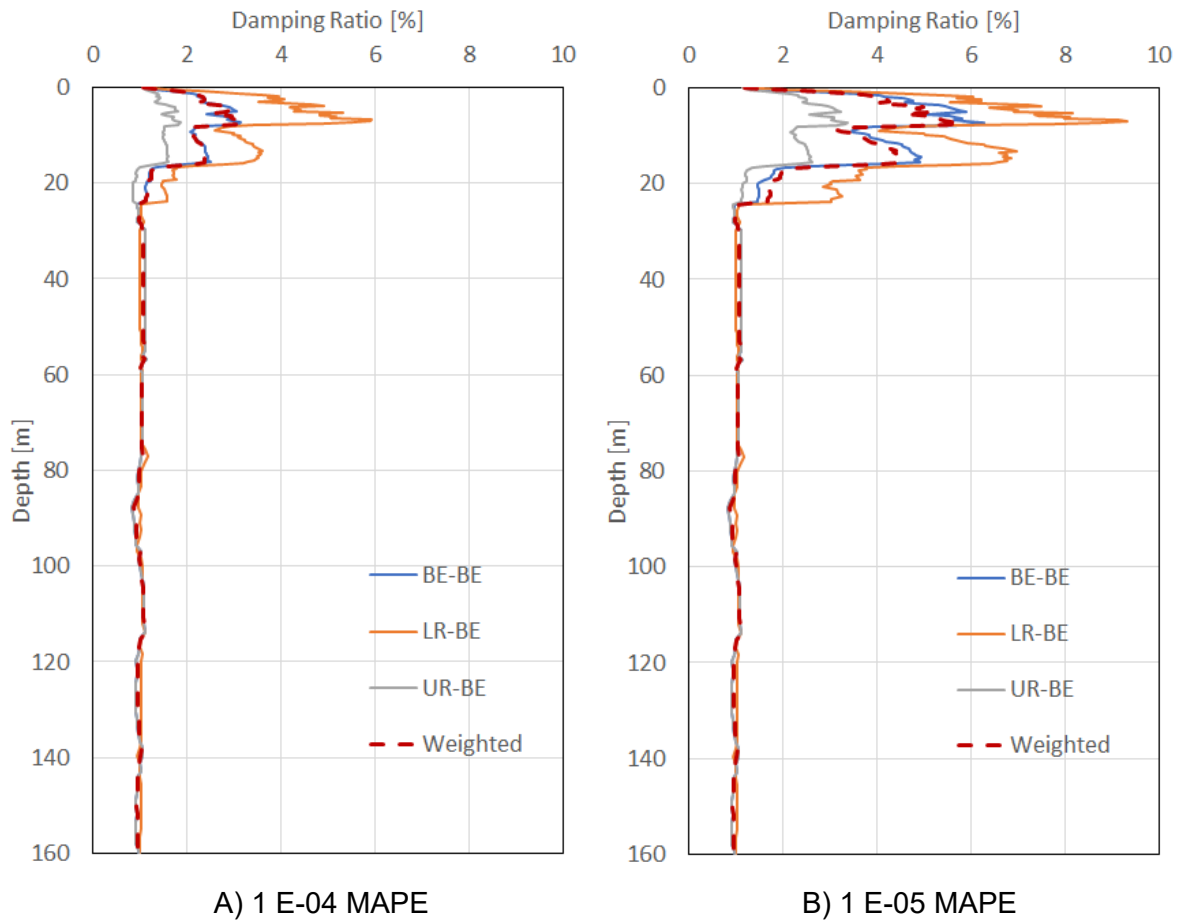


**Figure 3.3-20: Logarithmic Mean of Strain-Compatible Shear Wave Velocities**



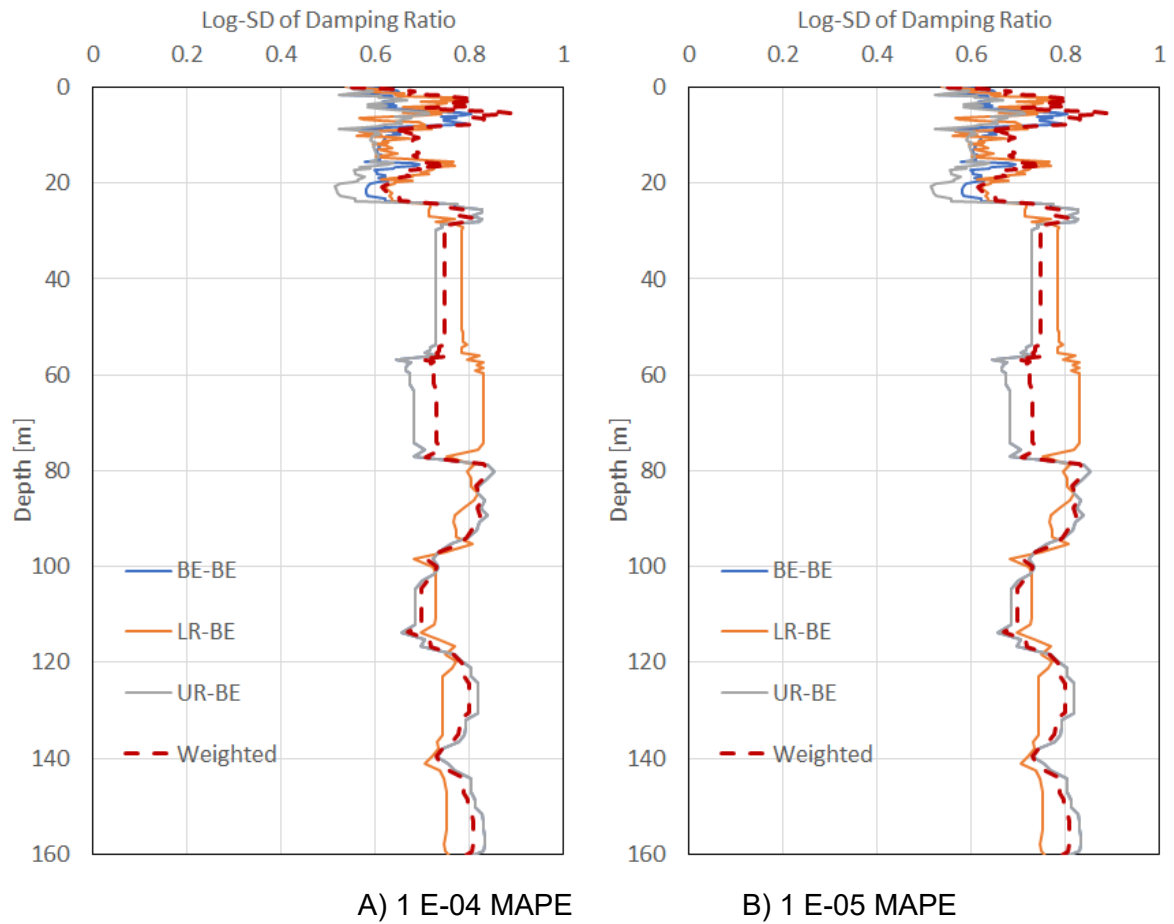
**Figure 3.3-21: Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocities**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



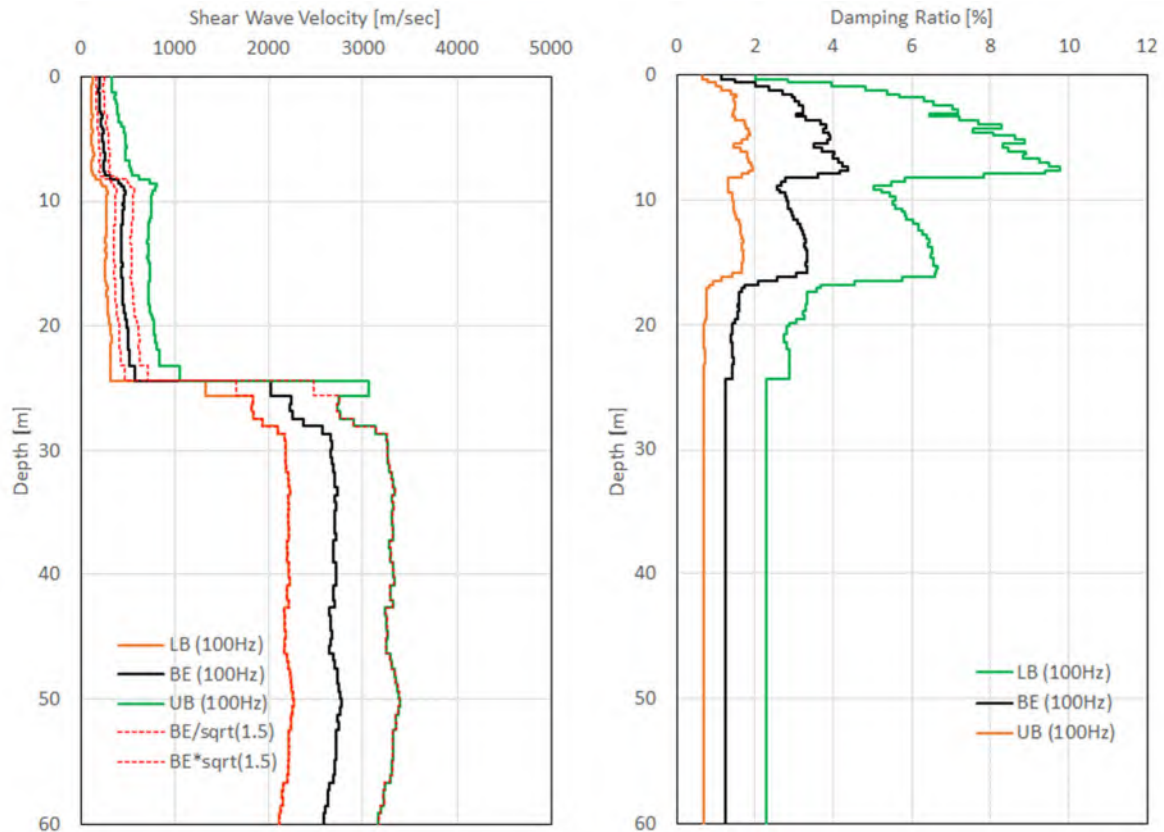
**Figure 3.3-22: Logarithmic Mean of Strain-Compatible Damping Ratios**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.3-23: Logarithmic Standard Deviation of Strain-Compatible Damping Ratios**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



a) Shear Wave Velocity

b) Shear Wave Damping Ratio

**Figure 3.3-24: Strain-Compatible Shear Wave Velocity and Damping  
Using 100 Hz Interpolation**

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

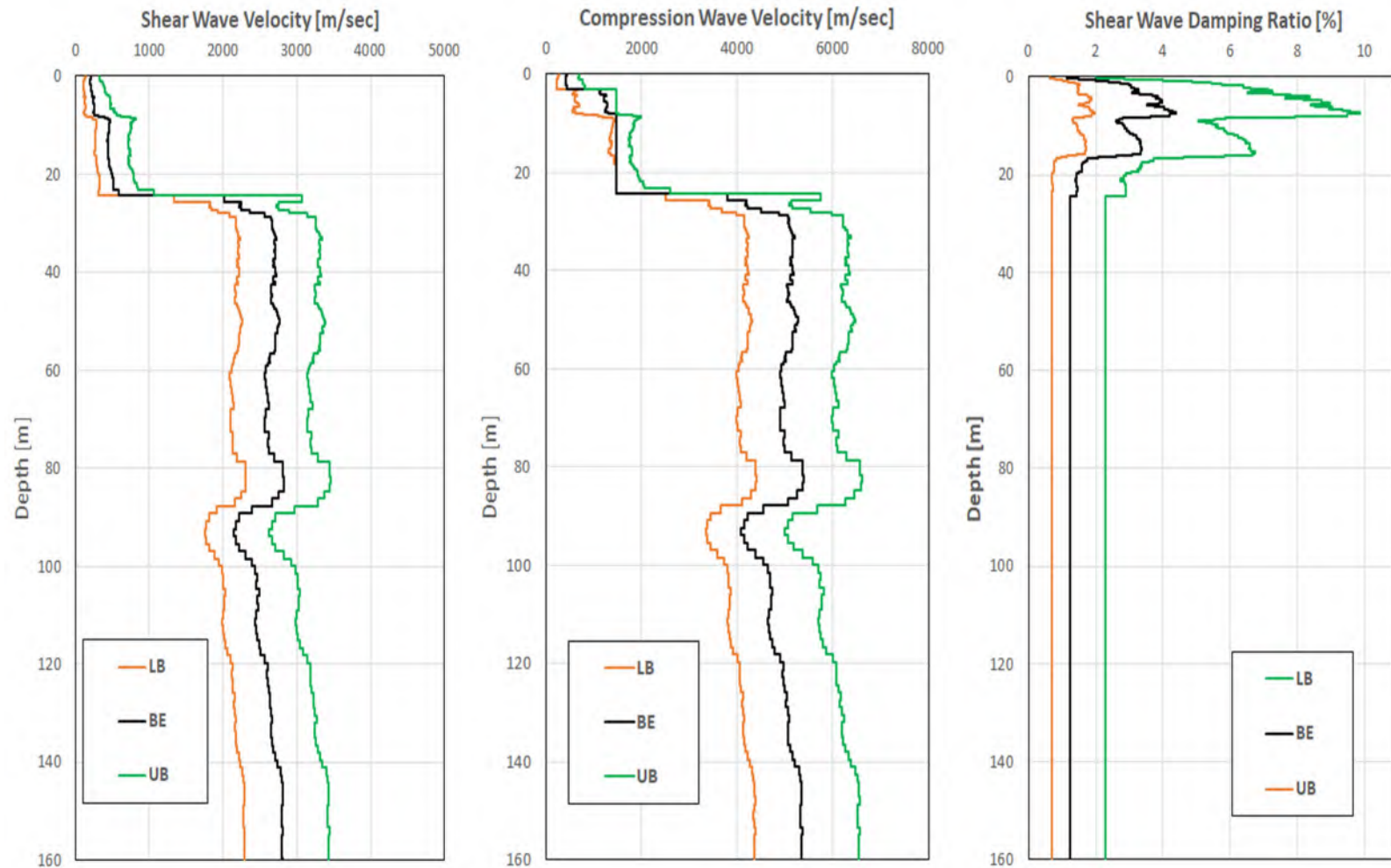


Figure 3.3-25: Subgrade Profiles for Bounding BWRX-300 Seismic Analyses

### **3.4 Protection Against Internal Hazards**

This section discusses design basis internal hazards that could compromise the safety functions of SC1 SSC and preventive, and mitigation measures implemented in the design to eliminate their adverse effects. SC2/SC3 SSC credited in the fault evaluation with mitigating fault sequences initiated by internal hazards are also protected against internal hazards. Forbdba internal hazards, refer to Chapter 15, Sections 15.5 and 15.6.

The list of internal hazards considered in the BWRX-300 design is generated from the industry guidelines and the specifics of the BWRX-300 technology. These hazards are in accordance with CNSC REGDOC-2.5.2 (Reference 3.4-1), Section 7.4.1 supplemented by IAEA SSG-64 (Reference 3.4-2), which supersedes IAEA NS-G-1.11 (Reference 3.4-3) referenced in CNSC REGDOC-2.5.2. Screening methodology of internal hazards for safety analysis purposes and ultimately confirmation of adequacy of protection measures is identical to that of the external hazards presented in Section 3.3.

Protection and mitigation methods considered in the design are in line with the design safety objectives and D-in-D concept discussed in Subsections 3.1.1 and 3.1.6, respectively. They include the use of separation, barriers/shielding and monitoring programs as described in Subsection 3.1.5 to preclude unacceptable radiation releases following accidents due to internal hazards.

When applicable, loads generated by internal hazards are considered in the BWRX-300 design in compliance with requirements in Section 7.15.1 of CNSC REGDOC-2.5.2 and CSA N291 (Reference 3.4-4). Combination of loads from randomly occurring individual internal hazards is also considered in the design to ensure structure are adequately protected against internal hazards.

#### **3.4.1 Internal Fires, Explosions and Toxic Gases**

Protection and mitigation measures considered in the BWRX-300 design against internal fires, explosions, and toxic gases to comply with CNSC REGDOC-2.5.2, Section 7.4.1 are discussed in Subsections 3.4.1.1 through 3.4.1.3.

##### **3.4.1.1 Internal Fires**

Protection against internal fires is provided by:

1. A fire protection system to detect, notify, and suppress internal fires and the implementation of a comprehensive fire protection program.
2. Designing, locating, and compartmentalizing SSC to minimize the probability and effect of fires and explosions. Separation is provided between defense lines to the extent that defense lines are credited in the fault evaluation to mitigate the same event. Separation is provided using passive fire barriers to subdivide the plant into separate areas. Separation also confines the effects of fires to a single compartment or area minimizing the potential for adverse effects from fires on redundant SSC.

The fire protection system comprises fire alarms, automatic fire suppression, smoke removal, yard fire main with hydrants, building standpipe and hose stations, fire pumps, water supply and fire extinguishers. Details including design features and parameters of the fire protection system are provided in Chapter 9A, Section 9A.6.

The comprehensive fire protection program covers administrative controls, procedures, periodic inspections, maintenance, testing and training of personnel to ensure a safe shutdown of the plant and the health and safety of plant operators and the public. This program ensures the following life safety performance objectives are met during all operational modes and plant configurations:



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Fire hazard controls are included in design and operational stages
- Fire notification means are provided
- Safe egress and/or areas of refuge are provided for occupants for use in the event of a fire
- A safe environment and other required support are provided for essential staff so they can perform all necessary plant control functions during and following a fire
- Protection for personnel performing emergency services is provided both during and following a fire
- Access and emergency lighting are provided for all areas where manual firefighting, evacuations, or operation field actions are expected

The fire safety assessments form a key element in the fire protection program. The fire safety assessments document a systematic review of the fire hazards at DNNP and the potential consequences of design basis fire events.

To satisfy requirements in CSA N293 (Reference 3.4-5) and CSA N293S1 (Reference 3.4-6), a fire hazard assessment is performed as discussed in Chapter 9A, Subsection 9A.6.10 to identify the specific fire hazards and fire protection capabilities for the plant. Chapter 9A, Subsection 9A.6.10 also discusses the fire safe shutdown analysis that evaluates fire effects on the safe shutdown systems to demonstrate compliance to the related requirements of the CSA N293 standard. Methodology for these evaluations is illustrated in Chapter 9A, Figures 9A.6.10-1 and 9A.6.10-2.

The BWRX-300 fire protection design satisfies requirements in CSA N293, CSA N293S1 and the applicable clauses of the NBC (Reference 3.4-7). The D-in-D principle discussed in Subsection 3.1.6 is used to achieve a high degree of fire protection by providing redundancy, diversity and balance in the fire protection measures included in the design to prevent, detect, suppress, and limit the effects of fires. A summary of fire protection measures for the Power Block buildings is provided in Subsections 3.4.1.1.1 and 3.4.1.1.2. Fire protection design features are discussed in Chapter 9A, Section 9A.6 and Chapter 9B, Sections 9B.2 and 9B.3.

#### **3.4.1.1.1 General Protection Measures for Power Block Building Structures**

The Power Block buildings are generally steel frame construction except for the RWB and the TB portion enclosing the main steam line which are of reinforced concrete construction, and the RB which is constructed using Steel Bricks™. To satisfy requirements in Section 7.12.1 of CNSC REGDOC-2.5.2, the walls, floors, and ceilings are designed to have 3-hour fire resistance ratings where required based on high combustible loadings (lubrication oil tank, for example) in the room or where an adjacent room contains equipment or systems from a different safety class division.

Corridors, stair enclosures and elevator hoistways that do not communicate between areas of different safety class divisions may have walls with a 2-hour minimum fire rating. Non-concrete interior walls are constructed of metal studs and gypsum wallboard to the required fire resistance rating.

Doors, including frames and hardware, penetrating rated fire barriers comply with the NBC or equivalent National Fire Protection Association (NFPA) ratings for that barrier.

The fireproofing of structural steel members where required by calculation based on combustible loading, is accomplished by application of an Underwriters Laboratory (UL) of Canada or equivalent UL - listed or Factory Mutual approved cementitious or ablative material, or by UL -

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

listed or Factory Mutual approved boxing design. The required fire rating determines the fireproofing material thickness.

To satisfy requirements in Section 6.8.1.4 of CSA N293, wall and ceiling surface finishes are specified to meet flame spread index of 0-25 and smoke-developed index of 0-100 in accordance with CAN/ULS-S102 (Reference 3.4-8). Floor finishes have a flame spread rating of 0-300 and a smoke development classification less than 450 when tested in accordance with ASTM E648 (Reference 3.4-9) and ASTM E662 (Reference 3.4-10).

Suspended ceilings, including the lighting fixtures are of non-combustible construction in accordance with Section 5.7.1.1 of CSA N293.

To prevent the spread of spilled flammable and combustible liquids, including contaminated firefighting water, diking, draining or a combination of both is used to contain and control the volume of liquids in the buildings. Spill control measures are also included in the design to contain the contents of any above grade oil-filled vessel or tank larger than 208 liters and all tanks containing chemicals used in water/wastewater treatment or quality control.

#### **3.4.1.1.2 General Protection Measures for Systems and Components**

Complying with Section 6.8.4.1 of CSA N293, the BWRX-300 design minimizes the use of plastics, wood and other combustible materials in electrical equipment, cable raceways and wiring racks. Non-combustible and heat-resistant materials are used wherever practical throughout the unit.

Electrical cable in open tray raceways is limited to low voltage cable and meets IEEE 383 standards (Reference 3.4-11) in accordance with Section 6.8.4.4 of CSA N293. Vertical cables have a maximum vertical char of 1.5m when tested in accordance with the vertical flame tray test (Method 2-FT4) test in CSA C22.2 No. 2556 (Reference 3.4-12). Circuitry over 1000 volts is in conduit.

Certain areas of the plant have cable trays in stacked array. Where stacking of trays occurs, power cable, which is the most susceptible to internally generated fires, is routed in the uppermost tray to the greatest extent possible to provide isolation from other trays in the stack. A vertical separation is provided between horizontal cable trays. Groups of stacked trays for redundant SCN cables are separated horizontally.

Piping and cable tray penetrations are provided with fire-stops when penetrating fire rated barriers in accordance with Section 6.5.2.1 of CSA N293. Electrical cable fire-stops are tested to demonstrate a fire rating equal to the rating of the barrier they penetrate in accordance with Section 6.5.2.1 of CSA N293. As a minimum the penetrations meet the requirements of NUREG-1552 (Reference 3.4-13), including Supplement 1 of CSA C22.2 No 0.3 (Reference 3.4-14). The tests are performed or witnessed by a representative of a qualified, independent testing laboratory. The documented test results for the acceptable fire-stops are made a part of the plant design records.

To satisfy requirements in Section 6.3.1.1 of CSA N293, control, power, or instrument cables and equipment of redundant systems used for achieving and maintaining safe shutdown, are separated from each other by three hour rated fire barriers, except within inerted containment. Where the equipment of more than one division is required to be located within a single fire area (Control Room), cables are within conduit or a floor trench.

Fire separations are required to separate redundant fire safe shutdown systems and separate safe shutdown systems from other hazards.

Suitable design of the ventilation systems limits the consequences of a fire by preventing the spread of the products of combustion to other fire areas. Means are provided to ventilate,

exhaust, or isolate the fire area as required, with consideration given to the consequences of ventilation system failure caused by the fire, resulting in a loss of control for ventilating, exhausting, or isolating a given fire area.

Filter media (excluding charcoal filters and High Efficiency Particulate Air (HEPA) filters) used in air handling systems meet the combustibility requirements of Class I in accordance with CAN/ULC-S111(Reference 3.4-15).

HVAC penetrations through 2-hour or 3-hour rated fire barriers are provided with fire/smoke dampers compatible with the rating of the fire barrier.

In accordance with Section 6.8.4.2 of CSA N293, electrical cabinets are designed to limit flame spread across cabinets.

#### **3.4.1.2 Internal Explosions**

The BWRX-300 fire hazard assessment evaluates the combustible loading along with the associated suppression requirements for each of the Power Block significant rooms and document the findings on the room data sheets.

Potential explosions of the following components are considered in the design:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen fires
- Offgas/hydrogen recombiners
- Transformers
- Transient combustibles
- Turbine auxiliaries

To satisfy requirements of CNSC REGDOC-2.5.2, Section 7.4.1, separation is provided between defense lines to the extent that defense lines are credited in the fault evaluation to mitigate the same event. Design measures considered include the use of fire barriers and blowout doors where flammable and combustible materials are located, and redundancy to enhance the reliability of systems.

Non-combustible and heat-resistant materials are also used, wherever practical throughout the Power Block, particularly in locations such as the containment and control rooms to reduce the risk of fires and explosions.

Administrative controls are also implemented to ensure stored chemicals and combustibles cannot ignite or react in sufficient quantities to impact nuclear safety. Collapse of structures, pipe whip, jet effects, and internal flooding as a result of internal explosions is also considered in the design.

#### **3.4.1.3 Release of Internal Hazardous (Toxic) Gases**

Plant personnel are protected from the adverse effects due to uncontrolled release of hazardous substances as a result of fires or internal explosions in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1, 7.12.1 and 7.12.2.

Preventive and mitigation measures against the release of hazardous and toxic gases include a proper design of ventilation systems to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Refer to Chapter 9A, Sections 9A.5 and 9A.6 for details of the BWRX-300 HVAC and fire protection systems, respectively.

Complying with CNSC REGDOC-2.5.2, Sections 8.10.1 and 8.10.2, the habitability of the MCR and SCR is ensured by designing the HVAC systems in these rooms to detect and limit the introduction of airborne radioactivity, toxic gas or smoke into the rooms as described in Chapter 6, Section 6.4. As stated in Chapter 6, Section 6.4.2.1, habitability requirements in the control rooms are maintained without credit for any breathing apparatus or protective clothing.

HVAC systems also supply outside air into the SCCV via the containment inerting system and exhaust inerting gases to provide a habitable environment for maintenance personnel during outage and maintenance periods.

### **3.4.2 Internal Flooding**

SC1 SSC and SC2/SC3 SSC credited with flood event mitigation in the fault evaluation are protected against internal flooding in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1.

Appropriate means are included in the design to prevent failure of SSC that are not designed to be submerged or exposed to spray as a result of flooding. They include the use of redundant system trains or divisions, structural barriers or compartments, curbs and elevated thresholds, and a leak detection system.

The design of the integrated RB structures considers the loads associated with the post-accident internal flooding of the containment following a DBA. The hydrostatic loads from the maximum possible water level are applied as pressures to the affected walls and mat foundation and applicable loads are also used for design of containment metal components.

The BWRX-300 internal flooding analysis identifies flooding sources, equipment in each area, and maximum internal flood levels in each area. The sources of internal flooding hazards include:

- Leaks and breaks in pressure retaining components
- High-energy piping breaks and cracks
- Moderate-energy piping through-wall cracks
- Pump mechanical seal failures
- Failure of isolating devices
- Storage tank ruptures
- Actuation of fire protection system
- Flow from upper elevations and nearby areas

The flood level in each internal area is determined by evaluating the inflow due to internal flooding sources, outflow from area compartment, and accumulation in each compartment area due to net flow.

### **3.4.3 Internal Missiles**

Complying with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1, the BWRX-300 design includes preventive and mitigation measures against internal missiles. The methodology used to

determine internal missiles is discussed in Subsection 3.4.3.1, while Subsection 3.4.3.2 provides the general preventive and mitigation measures considered in the design.

#### **3.4.3.1 Sources of Internal Hazards**

Potential missiles inside and outside containment and turbine missiles are identified, and their statistical significance determined. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or exceedance of radiological release limits. Criteria for determining statistically significant missiles are obtained from applicable portions of U.S. NUREG-0800 (Reference 3.4-16), SRP 3.5.1.1 through 3.5.1.3.

These missile sources could result from in-plant component overspeed failures or high-pressure system ruptures in compliance with CNSC REGDOC-2.5.2, Section 7.4.1. Rotating equipment failures include evaluations of pumps, fans, blowers, diesel generators, compressors, and turbines. Potential missiles from failure of pressurized components include valve bonnets, valve stems, pressure vessels, thermowells, retaining bolts, and blowout panels.

#### **3.4.3.2 Protection from Internal Missile Hazards**

Preventive and mitigative measures considered in the BWRX-300 design against internal missiles include the following:

- Locating the system or component in an individual missile-proof structure
- Physically separating redundant systems or components of the system from the missile trajectory path or calculated range
- Providing localized protection shields or barriers for systems or components
- Designing the particular structure or component to withstand the impact of the most damaging missile
- Providing design features on the potential missile source to prevent missile generation
- Orienting the potential missile source to prevent unacceptable consequences caused by missile generation

Refer to Subsection 3.3.5.4 for barrier design procedures for impactive loads, including internal missiles.

#### **3.4.4 Pipe Breaks**

BWRX-300 SC1 SSC and SC2/SC3 SSC credited with event mitigations in the fault evaluation are adequately protected from the consequences associated with a postulated rupture of high-energy piping and crack of moderate-energy piping inside and outside containment in compliance with Sections 7.4.1 and 7.7 of CNSC REGDOC-2.5.2 and IAEA SSG-64. Design bases and measures used to protect these SSC, referred to in the following subsections as essential SSC, are discussed in Subsections 3.4.4.1 and 3.4.4.2.

Effects that may result from a postulated rupture of high-energy piping include (1) pipe whipping, (2) pipe break reaction forces, (3) jet impingement forces, (4) blast waves, (5) sub-compartment pressurization, (6) decompression waves, (7) Missile generation, (8) environmental effects and (9) Flooding.

In the BWRX-300 design, a whipping pipe may hit a target and cause secondary failure in the target object depending on the thrust force, materials and sizes of the pipe/target. Severance in the target may occur and form a missile. A pipe whipping about a plastic hinge is not assumed to cause severance at the plastic hinge. Therefore, a break cannot cause the whipping pipe to act

as a missile. Criteria related to the evaluation of and protection against missiles, including those resulting from jet impingement or a whipping pipe, are provided in Subsection 3.4.3.

Protection against flooding and environmental effects as a result of high-energy pipe breaks are discussed in Subsections 3.4.2 and 3.9.4, respectively.

#### **3.4.4.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment**

##### **3.4.4.1.1 Design Basis**

In addition to meeting requirements in CNSC REGDOC-2.5.2 and IAEA SSG-64, the BWRX-300 pipe break event protection also conforms to 10 CFR 50 Appendix A (Reference 3.4-17), General Design Criterion 4. To supplement the guidance provided in IAEA SSG-64, the design bases for this protection are in compliance with NRC Branch Technical Position (BTP) 3-3 (Reference 3.4-18) and BTP 3-4 (Reference 3.4-19) included in Subsections 3.6.1 and 3.6.2, respectively, of U.S. NUREG 0800. BTP 3-4 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

1. Assure that the reactor can be shut down safely and maintained in a safe shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits with Loss of Preferred Power (LOPP).
2. Assure that containment integrity and leak tightness are maintained.

##### **3.4.4.1.2 Design Evaluation**

An analysis of pipe break events is performed to identify those essential systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the pipe break event.

Pipe break events involving high-energy fluid systems are evaluated for the effects of pipe whip, jet impingement, flooding, sub-compartment pressurization, and other environmental effects. Pipe break events involving moderate-energy fluid systems are evaluated for wetting from spray, flooding, and other environmental effects.

Adequate protection is provided against the effects of pipe break events for essential SSC to an extent that their ability to shut down the plant safely or mitigate the consequences of the postulated pipe failure is not impaired. This is accomplished by means of design features such as physical separation, jet shields and pipe whip restraints or by designing the SSC to accommodate applicable loads due to postulated pipe failure.

##### **3.4.4.1.3 General Protection Measures**

The direct effects associated with a particular postulated break or crack are mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the following specific measures for protection against actual pipe movement and other associated consequences of postulated failures:

1. Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

2. As an alternative to protective measures, SSC identified as essential targets under postulated pipe breaks are analyzed to show that the essential functionality remains available under all applicable loading conditions resulting from the pipe break.
3. The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.
4. Protection of SCN systems and components from the effects of postulated pipe breaks is considered where a resulting failure of the SCN system or component could lead to failure of an essential SSC. This includes consideration of coatings and insulation materials which could result in debris generation

### **Separation**

To meet requirements in CNSC REGDOC-2.5.2, Section 7.6.1.1, the plant layout arrangement provides physical separation and segregation of essential SSC to the extent practicable to provide sufficient distance such that the effects of the failure cannot impair their essential functionality.

Physical separation between redundant safety class systems supporting Defense Line 3 (DL3) with their related auxiliary supporting features is another basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

### **Pipe Whip Restraints**

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, analysis of the SSC or enclosures alone, and when it is necessary to limit the piping movement (pipe whip) following a postulated break. Restraints are located based on the specific postulated break locations determined in accordance with Subsection 3.4.4.2. After the restraints are placed, the piping and essential SSC are evaluated for jet impingement and pipe whip. For those cases where unacceptable jet impingement damage could still occur, barriers, shields, or enclosures are utilized in conjunction with pipe whip restraints.

The design criteria for restraints are given in Subsection 3.4.4.2.

### **Barriers, Shields, and Enclosures**

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present because of spatial separation or existing plant features, additional barriers, deflectors, shields, or guard pipes are provided as necessary to meet the functional protection requirements of essential targets.

Structures acting as barriers, shields, or enclosures are designed to withstand the consequences of postulated pipe failures (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with other internal hazards such as missiles and loadings associated with the DBE within their respective design load limits. Procedures used to design these structures are provided in Subsection 3.3.5.4.

The BWRX-300 barrier design ensures a resistance to impulsive loads that is at least 20% greater than the steady-state magnitude of the impulsive load in accordance with regulatory guidance of U.S. NRC RG 1.243 (Reference 3.4-20), Regulatory Position 11.1.2 and provisions of CSA N291, Clause A.3.5.1.

#### **3.4.4.1.4 Protective Features and Operator Actions**

All available systems are considered for mitigating the consequences of a failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such

as unit trip and LOPP, and of the assumed single active component failure and its direct consequences.

As stated in Chapter 15, Section 15.5, no operator actions are required to mitigate the effects of high-energy pipe breaks.

#### **3.4.4.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping**

This section discusses the location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate - energy fluid system piping inside and outside of the primary containment. This information provides the design basis for the requirements for protection of essential SSC.

##### **3.4.4.2.1 Criteria Used to Define Break and Crack Location and configuration**

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

#### **Definition of High-Energy Fluid Systems**

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 93.3°C; and
- Maximum operating pressure exceeds 1.9 MPaG.

#### **Definition of Moderate-Energy Fluid Systems**

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where either or both of the following are met:

- Maximum operating temperature is 93.3°C or less; and
- Maximum operating pressure is 1.9 MPaG or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

#### **Postulated Pipe Breaks and Cracks**

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are limited to postulation of cracks in piping and branch runs; these cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for conservative environmental conditions in a confined area where high and moderate-energy fluid systems are located.



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The following high-energy piping systems are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from damage effects:

- Main Steam
- Isolation Condenser System
- Control Rod Drive System
- Reactor Water Cleanup System
- Condensate Feedwater System
- Condenser Offgas System (in TB)

Moderate-Energy piping systems considered as potential candidates for a postulated pipe crack include the following:

- Boron Injection
- IC Pool Cooling
- Shutdown Cooling
- Fuel Pool Cooling
- Passive Containment Cooling
- Containment Inerting

#### **3.4.4.2.2 Location of Postulated Pipe Breaks**

Postulated pipe breaks are selected as follows:

##### **Piping in Containment Penetration Areas**

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated breaks. .

##### **ASME Code Section III Class 1 High-Energy Piping in Areas Other Than Containment Penetration**

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 1 piping (Reference 3.4-21) are postulated at the following locations in each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress range or fatigue usage values exceed the limits specified in BTP 3-4

##### **ASME Code Section III Class 2 and 3 High-Energy Piping in Areas Other Than Containment Penetration**

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 2 and 3 piping (Reference 3.4-22) are postulated at the following locations in those portions of each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress values exceed the limits specified in BTP 3-4

### **Non-ASME High-Energy Piping**

Breaks in seismically analyzed non-ASME high-energy piping systems are postulated according to the same criteria as for ASME Code Section III, Class 2 and 3 high-energy piping systems.

Breaks in non-seismically analyzed, non-ASME high-energy piping systems are postulated at each terminal end and at each intermediate location of potential high stress or fatigue, such as pipe fittings, valves, flanges, and welded-on attachments

#### **3.4.4.2.3 Location of Postulated Pipe Cracks**

Postulated pipe crack locations are selected as follows:

#### **Piping in Containment Penetration Areas**

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated cracks.

#### **High-Energy Piping in Areas Other Than Containment Penetrations**

With the exception of those portions of piping identified above as containment penetration areas, cracks in high-energy piping are postulated as follows:

1. For ASME BPVC Code, Section III Class 1 piping, at axial locations where the calculated stress range values exceed the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress values exceed the limits specified in BTP 3-4.
3. For piping which has not been evaluated to obtain stress information, through-wall cracks are postulated at axial locations that produce the most severe environmental effects.

#### **Moderate-Energy Piping in Areas Other Than Containment Penetrations**

With the exception of those portions of piping identified above as containment penetration areas, through-wall cracks in moderate-energy piping adjacent to safety class SSC are postulated except where:

1. For ASME BPVC Code, Section III, Class 1 piping the calculated stress range values are less than the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III, Class 2 or 3 and non-ASME class piping, the calculated stress values are less than the limits specified in BTP 3-4.

Through-wall cracks, unless the piping system is exempted above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.

Through-wall cracks are postulated in fluid system piping designed to non-seismic standards as necessary to assure that essential system and component functionality is maintained following a piping failure assuming a concurrent single active failure.

#### **Moderate-Energy Piping in Proximity to High-Energy Piping**

In cases where both high-energy and moderate-energy piping systems exist in a confined area, cracks are postulated in the piping system which leads to the more conservative environmental conditions.

#### **3.4.4.2.4 *Types of Breaks and Cracks to be Postulated***

##### **Pipe Breaks**

The following criteria are used to postulate breaks in high-energy fluid system piping at the identified locations:

1. For the purposes of considering dynamic effects, circumferential breaks are postulated only in piping having a nominal diameter greater than 25 mm.
2. Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than 100 mm.
3. Longitudinal breaks are not postulated at terminal ends.
4. Circumferential breaks are assumed at all terminal ends.
5. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsection 3.4.4.2.2, consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location is used to identify the most probable type of break based on the BTP 3-4 rules.
6. Where breaks are postulated to occur at each intermediate pipe fitting, weld attachment, or valve without the benefit of stress calculations, only circumferential breaks are postulated.
7. For a circumferential break, the dynamic force of the jet discharged at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient.
8. For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location.

##### **Pipe Cracks**

The following criteria are used to postulate through-wall leakage cracks in high- or moderate-energy fluid system piping at the identified locations:

1. Leakage cracks are only postulated in piping having a nominal diameter greater than 25 mm.
2. The postulated cracks are oriented circumferentially to result in the most severe environmental consequences.
3. Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half-pipe-diameter in length and one-half-pipe-wall thickness in width.
4. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.

#### **3.4.4.2.5 Analysis Methods to Define Blowdown Forcing Functions and Response Models**

##### **Analytic Methods to Define Blowdown Forcing Functions**

Analytical methods used to establish pipe rupture blowdown and jet thrust forcing forces are in accordance with ANSI/ANS 58.2 (Reference 3.4-23), Section 6.2.

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors.

Criteria used for calculation of fluid blowdown forcing functions include the following:

1. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).
2. For a circumferential break, the dynamic force of the jet discharge at the break location is based on the cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.
3. All breaks are assumed to attain full size within one millisecond after break initiation.

##### **Pipe Whip Dynamic Response Analysis Criteria**

Dynamic forces are assumed to cause pipe whip reaction whenever moments cause excessive plastic deformation and the formation of a plastic hinge. Significant motion occurs only when the thrust force acts through an arm of sufficient length to induce a plastic hinge. This length is called the plastic hinge length. When the stiffness of a piping system is such that a plastic hinge cannot form, the pipe lateral displacement is assumed to be equal to the pipe diameter.

Pipe whip restraints are used to prevent piping from deforming plastically by forming hinges. They absorb blowdown force energy and limit jet impingement's zone of influence.

The prediction of time dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from ruptured pipe is used as an input to evaluate the pipe whip dynamic response.

Pipe motion following circumferential breaks are assumed in the plane defined by the initial axis of the jet thrust force and rotation about a plastic hinge point, or at an intermediate point, such as the second change in direction, where the moment resisting capacity is less than straight pipe, provided the distance to this point is not significantly less than the plastic hinge length. The arc of the whipping pipe for planar motion is assumed to be limited to 180 degrees due to crimping at the plastic hinge and the pipe folding back against itself. Where a system consisting of piping, restraints and supporting structures is so complex that the assumption of planar motion is neither conservative nor realistic, the whip zone of influence can be conservatively enlarged to a region approaching a sphere with a radius equal to the distance between the break point and the first restraint. In lieu of this assumption, a more detailed elastoplastic analysis may be performed.

Longitudinal breaks in the form of axial split without pipe severance are postulated in the centre of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping configuration and produces out-of-plane bending.

Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

For restrained longitudinal breaks or those breaks for which it can be shown that the pipe resists bending elastically, the zone of whip influence is taken to be all points within a distance of one pipe diameter from the axis of the pipe, unless physically limited by piping restraints, structural members or piping stiffness. For unrestrained longitudinal breaks in elbow fittings, the out-of-plane forces are assumed to cause whipping through a zone of influence described by the rotation of the fitting through 360 degrees about an axis which connects the two plastic hinges formed in the attached legs of piping.

A whipping pipe is considered capable of rupturing impacted pipes of smaller nominal pipe diameter, and of developing through-wall cracks in impacted pipes of equal or larger nominal pipe sizes with thinner wall thickness.

If a whipping pipe contains a large in-line mass (such as a valve), or if there is a change in the pipe shape (e.g., an elbow) near the end of the pipe, rupture of target pipes which are equal to or larger than the whipping pipe is considered.

### **Pipe Whip Dynamic Response Methods**

Analytical models used to evaluate pipe whip dynamic response adequately represent the mass, inertia and stiffness properties of the piping system accounting for interaction effects of both the piping and pipe whip restraint.

Analytical methods used for piping response are based on those defined in ANSI 58.2, Section 6.3 and include complete system dynamic analysis, simplified dynamic analysis, quasi-dynamic analysis, energy balance analysis, and static analysis.

In cases where it is necessary to calculate stresses at locations which are far away from the break (e.g., in containment penetration break exclusion area), a more extensive model of the ruptured piping, supports, and pipe whip restraints is necessary.

If the snubbers or other seismic restraints are included in the piping model, they are modeled with the same stiffness used in the seismic analysis of the pipe. However, credit for seismic restraints cannot be taken if the applied load exceeds the ASME BPVC Code Section III (Reference 3.4-21, Reference 3.4-22 and Reference 3.4-24) Service Level D rating.

### **Pipe Whip Analysis Material Properties**

Strain rate effects and other material property variations are considered in the pipe whip analysis of piping and pipe whip restraints.

Material properties and design limits consistent with those stated in ANSI/ANS 58.2, Sections 6.6.2 and 6.6.3 are applied for plastic deformation design of piping and pipe whip restraint design under dynamic and steady-state loading conditions.

#### **3.4.4.2.6 *Dynamic Analysis Methods to Verify Integrity and Operability***

### **Jet Impingement Analyses and Effects on Essential Components**

For each postulated circumferential and longitudinal break, an evaluation of jet impingement effects on essential targets including jet impinging force, thermal energy, and moisture is completed in accordance with the methodology criteria in this section.

In the case of circumferential breaks, jets are assumed to be oriented axially with respect to the pipe. In the case of longitudinal breaks, jets are assumed to be oriented radially.

Potential targets, or portions of targets adjacent to the jet boundary, are assumed to be impinged upon when reasonable variations in jet geometry or pipe movement are considered.

In evaluating the potential for jet impingement on specific targets, consideration is given to the movement of the jet centreline due to pipe whip, including pipe-restraint interaction.

Thermal and moisture effects on essential targets are determined in accordance ANSI/ANS 58.2, Section 7.4 and 7.5.

Modeling of the jet geometry and determination of the jet impingement force acting on a target is calculated according to ANSI/ANS 58.2, Sections 7.2, 7.3, and Appendices C and D, with modifications applied as identified in NUREG/CR-7275 (Reference 3.4-25).

### **Pipe Whip Effects on Essential Structures, Systems and Components**

This section provides the criteria and methods used to evaluate the effects of pipe displacements on essential SSC following a postulated pipe rupture.

Pipe whip (displacement) effects on essential SSC can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurs in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays and conduits.

#### **(1) Pipe Displacement Effects on Components in the Same Piping Run**

Essential components located in the same run as the postulated break meet the applicable ASME Code class limits for Service Level D and limits to ensure required operability.

#### **(2) Pipe Displacement Effects on Essential Structures, Systems, and Components**

The criteria and methods used to calculate the effects of pipe whip on external components consist of the following:

1. The effects on barriers, shields, or enclosures credited for protecting essential SSC are evaluated in accordance with the barrier design procedures given in Subsection 3.3.5.4.
2. If the whipping pipe impacts an essential system or component, mitigating measures are established to ensure essential functionality is not lost for the postulated break scenario.

### **Loading Combinations and Design Criteria for Pipe Whip Restraint**

Pipe whip restraints are non-ASME code class components. As a result, other methods (i.e., testing) such as the use a reliable database may be used instead of the rules applied to ASME code class components for their design and sizing.

Pipe whip restraints are designed for both the thrust force at the pipe rupture location and the impact force of the pipe. The magnitude of these forces is a function of the pipe size, fluid temperature, and operating pressure.

Pipe whip restraints, as differentiated from piping supports, are typically designed only to function, and carry loads for an extremely low probability gross failure in a piping system carrying high-energy fluid. They are also required to remain functional following an earthquake up to and including the design basis DBE.

Pipe whip restraints are designed with sufficient clearances to prevent an increase in the pipe stresses by their presence during any normal mode of reactor operation or condition and are designed to allow for in-service inspection of the process piping with minimal obstruction.

#### **3.4.4.2.7 Analytic Methods to Define Blast Wave Interaction to SSC**

Sub-compartment pressurization due to postulated pipe breaks is considered where applicable.

#### **3.4.4.2.8 Sub-compartment Pressurization**

As discussed in Chapter 6, Subsection 6.3.2.2, the BWRX-300 containment sub-compartments do not contain large high-energy pipes and are, therefore, not subject to sub-compartment pressurization loads. For breaks outside the containment, mass and energy releases into the sub-compartments are calculated as described in Chapter 15, Subsection 15.5.9.2. Pressurization of the sub-compartments of the reactor building is calculated using the GOTHIC code described in Chapter 15, Subsection 15.5.1.2. The GOTHIC model of the RB includes all sub-compartments of the RB as lumped parameter volumes, including all flow passages between the rooms. This includes all doors and blowout panels which may be closed normally but may open if a pressure differential develops between the sub-compartments.

#### **3.4.4.2.9 Decompression Waves**

3-D thermal hydraulic code TRACG (See Chapter 15, Subsection 15.5.1.2) generates pressure time history in the annular region between chimney/shroud and RPV due to acoustic decompression wave as a result of a pipe break. Generated time history is part of the inputs to RPV primary structural FE model along with jet impingement, jet reaction and pipe whip restraint loads inputs to determine dynamic effects on RPV components, RPV internals and nozzles/pipings attached to RPV.

### **3.4.5 Other Internal Hazards**

#### **3.4.5.1 Hard Object Impact**

Complying with CNSC REGDOC-2.5.2, Section 7.15.3 and IAEA SSG-64, the BWRX-300 design considers hard object impact loads resulting from the drop of heavy loads lifted and handled in areas where SSC required for safe shutdown of the plant are located.

Drops considered are those most likely to occur during the handling of plant equipment for maintenance or during spent fuel transfer operations. Other drops considered are drops as secondary effects of other internal hazards or external hazards discussed in Section 3.3.

In accordance with U.S. NRC RG 1.244 (Reference 3.4-26), the BWRX-300 heavy load is defined per the provisions of U.S. NUREG-0612 (Reference 3.4-27) as any load, carried in a given area after a plant becomes operational, that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Critical heavy load handling evolutions considered are those where inadvertent operations or equipment malfunctions, separately or in combination, could:

- Cause a release of radioactivity
- Cause a criticality accident
- Cause the inability to cool fuel within the reactor vessel or within the Fuel Pool
- Prevent a safe shutdown of the reactor

Measures considered to reduce the potential of heavy load drops in the RB meet the D-in-D guidelines in U.S. NRC RG 1.244 and Section 5.1 of US NUREG-0612. They include a proper plant arrangement, the implementation of a heavy loads program as part of the plant procedures and effective means of lifting and transporting heavy loads designed to satisfy the single failure proof guidelines of Section 5.1.6 of US NUREG-0612.

Chapter 9A, Subsection 9A.8.1 provides an overview of the BWRX-300 heavy load program which identifies all heavy loads lifted during operation of the plant and the safe travel paths determined for their lifting. This program also manages the safe execution of heavy load evolutions.

Chapter 9A, Subsection 9A.8.1 describes the various cranes and hoists used to lift and transport heavy loads and applicable guides and standards used for their design. The RB polar crane main and auxiliary hoists meet the requirements of single failure proof systems in accordance with ASME NOG-1 (Reference 3.4-28). The refueling platform main hoist meets the requirements of a single failure proof hoist. Periodic inspection and maintenance of cranes are also planned to ensure their safe functioning.

#### **3.4.5.2 Failure of Non-Structural Element**

The failure of non-structural elements is considered in the BWRX-300 design.

Staircases and elevator shafts are evaluated and designed for interaction with plant Seismic Category A or B SSC in the event of DBE.

Architectural components and shielding blocks whose failure or dislocation could affect the safe operation of any Seismic Category A or B SSC are also evaluated for seismic interaction.

Scaffolding and other temporary structures considered a temporary alteration in support of maintenance are evaluated for seismic interaction as well, following the plant temporary structures procedure.

#### **3.4.5.3 Electromagnetic Interference**

Internal electromagnetic interference is caused by induction or radiation from installed equipment.

Complying with CNSC REGDOC-2.5.2, Section 7.5, safety class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

Qualification requirements for protection against electromagnetic interference are presented in Subsection 3.9.5.

Plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements are discussed in Chapter 8, Section 8.6.

#### **3.4.6 References**

- 3.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.4-2 IAEA Safety Standards Series No. SSG-64, "Protection against Internal Hazards in the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.4-3 IAEA NS-G-1.11, "Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.4-4 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.4-5 CSA N293, "Fire Protection for Nuclear Power Plants," CSA Group.
- 3.4-6 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.
- 3.4-7 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.4-8 CAN/ULC-S102, "Method of Test for Surface Burning Characteristics of Building Materials and Assemblies," Underwriters' Laboratories of Canada.



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.4-9 ASTM E648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source," American Society for Testing and Materials.
- 3.4-10 ASTM E662, "Standard Test Method for Specific Optical Density of Smoke Generated by Solid Materials," American Society for Testing and Materials.
- 3.4-11 IEEE 383-2015, "IEEE Standard for Qualifying Electrical Cables and Splices for Nuclear Facilities," Institute of Electrical and Electronic Engineers.
- 3.4-12 CAN/CSA C22.2 No. 2556, "Wire and Cable Test Methods," CSA Group.
- 3.4-13 USNRC NUREG-1552, "Fire Barrier Penetration Seals in Nuclear Power Plants."
- 3.4-14 CAN/CSA C22.2 No 0.3-09, "Test Methods for Electrical Wires and Cables," CSA Group.
- 3.4-15 CAN/ULC-S111-13, "Standard Methods of Fire Tests for Air Filter Units," Underwriters' Laboratories of Canada.
- 3.4-16 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,"
- 3.4-17 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."
- 3.4-18 USNRC BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."
- 3.4-19 USNRC BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."
- 3.4-20 USNRC Regulatory Guide 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments."
- 3.4-21 ASME BPVC-III NB, "Section III - Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 3.4-22 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 3.4-23 ANSI/ANS 58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," American National Standards Institute/American Nuclear Society.
- 3.4-24 ASME BPVC-III NE-2021, "BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components," American Society of Mechanical Engineers.
- 3.4-25 USNRC NUREG/CR-7275, "Jet Impingement in High-Energy Piping Systems."
- 3.4-26 USNRC Regulatory Guide 1.244, "Control of Heavy Loads at Nuclear Facilities."
- 3.4-27 USNRC NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
- 3.4-28 ASME NOG-1, "Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.

### **3.5 General Design Aspect for Civil Engineering Works of Seismic Category Buildings and Civil Engineering Structures**

This Section presents the design principles, design basis requirements, criteria and applicable codes and standards used in the design of the BWRX-300 civil structures, including their foundations in compliance with requirements in CNSC REGDOC-1.1.2 (Reference 3.5-1), Section 4.5.5.

Below are the key PSAR sections that impact the BWRX-300 Civil/structural design that should be reviewed along with this section:

- Chapter 1 which provides the DNNP general site and facility layout, a description of the BWRX-300 buildings, plant operational modes, principles of safety management and applicable codes & standards utilized in the design
- Chapter 2 which described the characteristics of the DNNP site on which the BWRX-300 facility is constructed
- Chapter 3, Section 3.1, which provides the general design aspects and D-in-D safety framework utilized in the BWRX-300 design
- Chapter 3, Section 3.2, which provides the general classification of BWRX-300 SSC and the approach used to establish these classifications
- Chapter 3, Sections 3.3 and 3.4, which provide methodology and general design requirements for protection against the effects of external and internal hazards
- Chapter 9B which provides specific information on compliance with the design rules for civil engineering works and structures

From the site layout presented in Chapter 1, Appendix A, Figure A1.4-1, the primary buildings in the BWRX-300 Power Block consist of the Reactor Building (RB) which houses the containment, Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), and Reactor Auxiliary Bay. In the following sections, reference to the integrated RB structure is inclusive of the RB, containment, and containment internal structures, whereas RB is used to refer to the part of the integrated structure located outside of containment.

The seismic categorization of these structures is provided in Table 3.3-1. Per Subsection 3.2.3 and Table 3.3-1, the Seismic Category A integrated RB housing SC1 SSC has the utmost importance to safety and is credited for the safety analysis of the BWRX-300. RWB structures that support and protect equipment and components for storage and processing of highly radioactive gas, liquids and solid materials are categorized as RW-IIa. The CB, TB and Reactor Auxiliary Bay categorized as Non-Seismic structures are not credited in the safety analysis but are relied upon for their D-in-D function since they house and protect SC2 or SC3 systems and components. The RWB, CB, TB, and Reactor Auxiliary Bay can also affect the BWRX-300 safety considering their proximity to and interaction with the integrated RB structure.

Other civil structures for which design basis requirements are provided are the Pumphouse/Forebay structures and tunnels that support the condenser cooling and plant cooling water systems, and the Fire Pump Enclosure. For the location of these structures, refer to Chapter 1, Appendix A, Figure A1.4-1.

In accordance with Section 3.1 of CNSC REGDOC-1.1.5 (Reference 3.5-2) and Section 5.4 of CNSC REGDOC-3.5.3 (Reference 3.5-3), design principles for BWRX-300 structures are provided in a graded manner commensurate to their importance to safety. The primary focus of this Section is for the Seismic Category A integrated RB. Design principles for the RWB, CB, TB,

Reactor Auxiliary Bay, Pumphouse/Forebay and Fire Pump Enclosure structures are provided in Chapter 9B, Section 9B.3.

Remaining plant structures shown in Chapter 1, Appendix A, Figure A1.4-1 are not covered since they are not credited in the safety analysis.

### **3.5.1 General Design Principles for Seismic Category A Structures**

The BWRX-300 Seismic Category A integrated RB structure is designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, Anticipated Operational Occurrence (AOO) and DBA in compliance with requirements in CNSC REGDOC-2.5.2 (Reference 3.5-4), Sections 7.15.1 and 7.7. The robustness of the design to prevent potential release of radioactivity to the public and environment under Design Extension Condition (DEC) is considered in compliance with requirements in CNSC REGDOC-2.5.2, Sections 7.7 and 7.15.1 and is discussed in Subsection 3.5.6.

The integrated RB structure and its common foundation are primarily constructed using an advanced steel-plate composite system called Steel Bricks™. The Steel Bricks™ system has a configuration similar to the typical steel-plate composite system except that the tie-rods in the typical steel-plate composite system are replaced by diaphragm plates created by bending the plates that facilitates the fabrication process. The Steel Bricks™ modules used to construct the integrated RB comprise of a pair of steel faceplates, shear connectors, diaphragm plates, and concrete fill. The faceplates and concrete fill act as the composite system to provide strength and stability to the Steel Bricks™ system. The shear connectors facilitate the composite action between the faceplates and concrete fill, and the diaphragm plates act as shear reinforcement besides holding the system together. The design of the structures serving as the containment pressure boundary is performed in accordance with the provisions of ASME Boiler and Pressure Vessel Code (BPVC) as described in NEDC-33926P (Reference 3.5-5). The Steel-plate Composite Containment Vessel (SCCV) is designed in accordance with NEDC-33926P, as described in Subsection 3.5.3.1.

Similarly, the Class MC containment metal components are designed in accordance with the provisions of ASME BPVC, Section III, Division 1, Subsection NE (Reference 3.5-6).

ANSI/AISC N690 (Reference 3.5-7) that has been endorsed by U.S. NRC RG 1.243 (Reference 3.5-8), along with NEDC-33926P provide the specifications for the design, fabrication, construction, examination, and inspection of RB Steel Bricks™ and steel structures that do not provide the containment pressure boundary and for the containment internal structures.

These U.S. codes and standards are adopted for the BWRX-300 steel-plate composite structures (Steel Bricks™) since there are no equivalent standards or regulatory guidance in Canada.

Clause 6.1.2 of CSA N291 (Reference 3.5-9) permits the use of alternate design methods for design of nuclear structures and concrete containments in Canada. Requirements for design, fabrication, construction, examination, and testing of containment, containment internal structures, RB, and their foundations presented in Subsections 3.5.2 through 3.5.5 ensure compliance to the regulatory requirements in CNSC REGDOC-2.5.2 and meet the intent and ensure a level of safety and performance commensurate with the applicable Canadian standards.

#### **3.5.1.1 Structural Analysis Criteria for Seismic Category A Structures**

In accordance with requirements in CNSC REGDOC-1.1.2, Section 4.5.5 and CNSC REGDOC-2.5.2, Sections 7.13.1, 7.15.1, 7.22 and 8.6, the RB, containment and the containment internal structures are analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine structural design demands resulting from various design loads and design

load combinations. Evidence of qualification of these computer programs, including a description of the programs and extent of use, is presented in Appendix 3B.

The following Finite Element (FE) analyses are performed to obtain stress demands for the design of the BWRX-300 RB, containment, and containment internal structures:

- 1-g static SSI analyses
- Static and quasi-static analyses
- Thermal stress analyses
- Seismic SSI analyses

Static analyses provide design demands on the RB integrated structures from dead loads, live loads, earth pressure loads, hydrostatic and hydrodynamic loads, severe and extreme environmental loads, plant operating loads during normal operation, testing and abnormal plant conditions. Thermal analyses provide stress demands due to normal operating and accidental load conditions. Design Basis Earthquake (DBE) seismic demands are obtained directly from the results of one-step approach SSI seismic analyses discussed in Subsection 3.3.1.2.

The effect of interaction with the surrounding subgrade is incorporated in the analyses of the deeply embedded integrated RB by considering the surrounding soil and rock as a layered half-space continuum. The geotechnical design parameters used as input for the static and thermal analyses are developed as described in Subsection 3.5.2.2.

#### **3.5.1.1.1 FE Model of Integrated RB Structure**

To determine internal forces resulting from various loads and loading combinations, a detailed structural model is developed for the integrated RB, containment, and containment internal structures, including their foundations, penetrations, and openings, following the general FE modeling guidelines for the integrated RB structure discussed in Subsection 3.3.1.2 and NEDO-33914 Revision 2 (Reference 3.5-10), Section 5.1.1. The integrated structural FE model adequately represents the RB structural configuration for all main structural members and meets the mesh refinement and quality attributes required for calculation of structural stress demands. The use of the common model enables the FE results obtained from the different analyses to be directly combined in design load combinations per governing design codes.

Materials properties assigned to the integrated RB model depend on the analyzed loads and resulting stress responses. Unit weight properties are assigned to the models used for the 1-g static SSI analyses to adequately simulate gravity and earth pressure loads. The dynamic model of the integrated RB used for the seismic SSI analyses is assigned seismic mass inertia properties as discussed in Subsection 3.3.1.2.

As discussed in Subsection 3.3.1.2, stiffness properties are assigned to the SCCV and RB to reflect effective stiffness for load combinations without accidental thermal load. For load combinations with accidental thermal load, reduced stiffness is considered to account for the cracking effects on the redistribution of forces and moments. Spring elements are also used in the integrated FE element model to represent the stiffness of the connections between the different structural members that are designed to relieve stresses due to thermal expansion.

#### **3.5.1.1.2 1-g Static SSI Analyses**

Stress demands for the design of the integrated RB structure from dead loads and earth pressure design loads are obtained by applying the Earth gravity (1-g) load in the vertical direction to the SSI model described in Subsection 3.5.1.1. The 1-g static SSI analyses utilize the same substructuring method as the seismic SSI analyses described in Subsection 3.3.1.2. LB equivalent

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

linear stiffness properties and UB unit weight properties assigned to the subgrade model used in the analyses are discussed in Subsection 3.5.2.2.

Maximum dynamic responses of the SSI system that are equivalent to its static response under 1-g gravity load are calculated by applying on the 1-g SSI analyses model an equivalent static 1-g excitation in the vertical direction as vertically propagating compression wave. To simulate 1-g excitation, a harmonic acceleration time history is used with:

- A low frequency equal to the analysis frequency increment, and
- An amplitude equal to the Earth's gravity (g).

The 1-g excitation is applied at control point located at the surface of the site free-field model.

Stress demands obtained from the one-step 1-g static SSI analyses include the effects of static earth pressures simulated by the interaction of the integrated RB structural model with the subgrade FE model. Shell elements at the surface of the subgrade are included in the SSI model to simulate the applicable overburden inertia loads from the surrounding Power Block foundations and other surcharge loads.

Contact springs are used at the interfaces of the RB structure with the surrounding subgrade as discussed in Subsection 3.3.1.2. In accordance with the FE modeling guidance in NEDO-33914 Revision 2, Section 5.1.1, the following stiffness properties are assigned to the contact springs in the models used for the 1-g static SSI analyses to provide UB lateral soil pressures on the RB below grade exterior walls:

1. The contact springs in the direction normal to the RB exterior walls are assigned properties representing UB stiffness conditions at the SSI interfaces.
2. The friction at the RB exterior walls is not considered by assigning very low stiffness properties to the contact springs in vertical and tangential direction.

Results obtained from these contact spring elements serve for calculation of earth pressures on the below grade RB shaft exterior wall and mat foundation.

#### **Subgrade Modeling Assumptions for Deeply Embedded RB**

Per NEDO-3914, Section 5.1.2, the following assumptions related to the modeling of the subgrade are introduced in the 1-g Static SSI analyses to enable an efficient calculation of stress demands on the RB structure due to pressure loads from soil and rock surrounding and supporting the RB shaft:

1. The properties of the subgrade materials are represented by linear elastic constitutive models
2. The non-linearities at soil-structure interfaces are not considered
3. The rock mass is assumed continuous and the presence of cavities, fracture zones, joints, bedding planes, discontinuities and other weak zones is not considered

The soil and rock strata in the 1-g static SSI models are modeled based on the principles of continuum mechanics using isotropic linear elastic properties. Possible fracture zones, joints, bedding planes, discontinuities and cavities in the rock are not explicitly included in the design SSI analyses models. Bounding properties assigned to the soil and rock materials are discussed in Subsection 3.5.2.2.

The effects of non-linearities at soil-structure interfaces are addressed by using elastic contact spring stiffness properties that provide bounding structural demands.

Rock with disadvantageous fracture zones, joints, bedding planes and discontinuities is reinforced to create a more self-supporting rock mass. If needed, rock reinforcements are provided as initial ground support. The rock reinforcements and any other support provided during the excavation and construction may degrade and is inaccessible after construction. Therefore, the design addresses the rock loads remaining after the initial ground support degrades by including the potential weight of the rock in the static 1-g SSI analysis or by applying additional pressures on the RB outer shaft wall. Additional horizontal pressure loads are also applied on the model to account for possible residual stresses in the DNNP rock mass.

### **RB Design Earth Pressure Load Validation**

Validations of the earth pressure loads are to be performed following the guidelines in Section 5.1.3 of NEDO-33914 Revision 2 to ensure the 1-g SSI static analysis provides conservative earth pressure design demands on the deeply embedded RB structure.

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.13.1 and NEDO-33914 Revision 2, Section 4, Foundation Interface Analyses (FIA) are performed on models representative of the non-linear constitutive behavior of soil and rock materials surrounding the RB shaft and employ non-linear interface modeling features capable of capturing the effects of non-linearities at the subgrade structure contact surfaces. The results of the FIA are to be used for validation of the design earth pressures following the guidance of Section 5.1.3 of NEDO-33914 Revision 2.

#### **3.5.1.1.3 Static and Quasi-Static Load Analyses**

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1, the following static and quasi-static analyses are performed on the integrated RB FE model to calculate structural stress demands due to:

- Live loads
- Crane loads
- Structural Integrity Test (SIT) and accident condition containment internal pressure load including differential containment and RB sub-compartment loads
- Horizontal hydrostatic pressure loads on pool walls
- Groundwater pressure loads on the integrated RB common mat foundation and below-ground exterior wall
- Extreme wind and tornado loads on RB roof and exterior wall
- Rain and snow loads
- Seismic water sloshing and breathing mode quasi-static pressure loads on pool walls
- Quasi-static pressure High Energy Line Break (HELB) loads (jet impingement, blast loads)
- Equipment and pipe reaction loads including RPV reaction loads.
- Post-accident internal flooding loads

The analyses of global static and quasi-static loads that can affect the global response of the integrated RB consider the effect of subgrade stiffness. Following the sub-structuring methodology, design demands from these loads are obtained from subgrade stiffness impedance analyses performed on models consisting of two parts:

- Super-element representing LB stiffness of the subgrade surrounding the RB, and

- Integrated FE model of the RB, containment and containment internal structures described in Subsection 3.5.1.1.1.

The super-elements define the stiffness of the subgrade at the nodes of the RB interfaces with the surrounding soil. The stiffness properties of the super-elements are developed using a layered 3-D solid FE model. Subgrade stiffness properties assigned to the super-elements are described in Subsection 3.5.2.2. To adequately simulate half-space boundary conditions, the depth of these models is deeper than three times the largest foundation dimension. The horizontal extent of these models is more than three times the RB shaft diameter.

The nodes of the super-element are coincident with the nodes of the integrated RB FE structural model. The coincident super-element and structural model nodes are connected by contact spring elements as described in Subsection 3.5.1.1.2. LB stiffness properties are assigned to these contact spring elements to yield larger structural deformations and conservative design stress demands. Equivalent linear subgrade stiffness properties assigned for the subgrade stiffness impedance static analyses are discussed in Subsection 3.5.2.2.

Fixed bases analyses are performed for the local loads with smaller magnitudes that do not affect the Integrated RB mat common mat foundation or global response.

Demands due to hydrostatic lateral pressure loads are obtained from static analyses of the integrated RB model with vertical supports applied to all mat foundation nodes. Demands from the upward buoyant pressures on the mat foundation are obtained from a static analysis of the integrated RB structural model with vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat. The results from the two groundwater load analyses are enveloped and then combined with the results of the 1-g SSI analysis cases to obtain earth pressure and groundwater load demands for the design of integrated RB structure.

Additional Rock Pressure load analyses are performed to account for possible residual horizontal stresses in the DNNP rock strata. Two boundary conditions are considered for these analyses that result in conservative stress demands:

1. Vertical supports established at all mat foundation nodes and horizontal supports established at the central node of the mat; and
2. Vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat.

The results of these two sets of additional static rock pressures analyses are enveloped and then combined with the results of the 1-g SSI analyses to ensure the RB structural design adequately addresses the effects of anisotropic and heterogenous rock behavior and accounts for potentially unstable rock mass loads.

#### **3.5.1.1.4 Thermal Stress Analyses**

To calculate structural stress demands due to the normal operating and DBA temperature loads, sub-structuring thermal stress analyses are performed on the integrated RB FE structural model coupled with super-element representing UB stiffness of the subgrade.

Stiffness properties are assigned to the Steel Bricks™ shell elements to account for the stiffness reduction effects under normal operating and DBA temperature loads. The corresponding structural stiffness conditions are used for the analyses for design loads that occur in combination with the normal and accident thermal loads.

For the thermal analyses, UB stiffness properties are assigned to the super-element modeling the subgrade and to the contact elements modeling the soil-structure interfaces resulting in

conservative thermal stress demands for the design of the RB and containment structures. Equivalent linear subgrade stiffness properties assigned for the thermal stress analyses are discussed in Subsection 3.5.2.2.

### **3.5.2 Foundations**

This section presents general design rules for the common Steel Bricks™ mat foundation supporting the integrated RB structure. Design rules for other foundations are discussed in Chapter 9B, Section 9B.3.

#### **3.5.2.1 Applicable Codes, Standards and Other Specifications**

Applicable codes, standards and specifications for the containment and RB common Steel Bricks™ foundation are the same as those for the superstructures.

The jurisdictional boundary for the application of the NEDC-33926P to the containment is the portion within the perimeter or exterior surface of the SCCV as shown in Figure 3.5-1.

The jurisdictional boundary for application of the ANSI/AISC N690 to the non-pressure retaining portion of the common foundation is the portion spanning from the exterior surface of the SCCV to the exterior surface of the RB (See Figure 3.5-1).

#### **3.5.2.2 Bounding Subgrade Design Parameters**

Bounding subgrade parameters are determined based on data available prior to the completion of the complete characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative subgrade property inputs adequately address uncertainties related to the use of incomplete characterizations of the DNNP site geotechnical and seismic conditions.

Based on the information from the available groundwater flow patterns and conditions at the DNNP site provided in NK054-REP-01210-00011 (Reference 3.5-11) and NK054-REP-07730-00005 (Reference 3.5-12), an Upper Bound groundwater level at elevation 85 m CGD corresponding to a depth of 3 m below the plant grade at elevation 88 m CGD is considered a parameter for the bounding design.

The geotechnical and hydrological investigations of the DNNP site have been completed and bounding subgrade design parameters determined (see Chapter 2, Subsection 2.7.5). The data collected from ground water measuring wells at the DNNP site indicate an upper bound nominal water table at a shallower depth of 2 m. The increase of an additional meter in the nominal ground water table elevation results in a 6% higher magnitude of the total force from ground water pressure load than the one calculated using the bounding design ground water table at 3 m depth.

The exterior RB wall is the main structural member resisting the below grade lateral pressures applied on the RB integrated structures. These below grade lateral loads include the static earth pressure, ground water hydrostatic pressure, and additional rock pressure that account for a large majority of the demand on the below grade portion of the exterior RB wall in approximately equal shares. Therefore, the effect of the marginal 6% increase in the ground water pressure, that represents no more than a third of the total structural demand on the exterior RB, is negligible and well bounded by the available structural design margins (see Chapter 9B, Appendix 9B.G).

Identification and evaluation of potentially liquefiable cohesionless soil strata under the BWRX-300 Power Block structures is performed in accordance with CSA N289.3 (Reference 3.5-13) and in compliance with requirements of CNSC REGDOC-2.5.2, Section 7.15.1.



#### **3.5.2.2.1 Bounding Equivalent Linear Subgrade Static Profiles**

As described in Subsection 3.5.1.1, the structural design demands due to static earth pressures on the RB below grade exterior walls are obtained from the 1-g static analyses of the integrated RB FE model embedded in a layered half-space continuum model representing the surrounding soil and rock. To account for the interaction of the RB integrated structures with the surrounding subgrade, super-elements representing the stiffness properties of the layered subgrade materials are used in the static and thermal analyses, as described in Subsection 3.5.1.1.

The 1-g static SSI analyses, subgrade impedance analyses and thermal stress analyses use profiles of bounding equivalent linear soil and rock properties developed using information from the existing laboratory tests and in-situ measurements taken in the vicinity of the DNNP site and following the recommendations of NEDO-33914 (Reference 3.5-10), Section 5.2.1. They consist of:

- Effective unit weight that for soil materials below groundwater table are calculated as the total unit weight of soil minus the unit weight of water
- Elastic and shear Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions
- Soil and rock Poisson's ratios representative of at-rest lateral pressure conditions

The bounding equivalent linear subgrade static profiles reflect anticipated as-built conditions at the site after construction of the BWRX-300 SMR that include engineered fill from about elevation 80 to 82 m CGD to the final grade at elevation 88 m CGD. The layering of the engineered fill, in-situ soil and rock materials in these bounding subgrade static profiles corresponds to the layering of dynamic subgrade properties described in Subsection 3.3.1.1.1 that are used as input for the DNNP site-specific seismic analyses.

Bounding static soil properties of in-situ soil materials are determined based on the results of in-situ tests and laboratory test results presented in the 2012 NK054-REF-01210-0418696 (Reference 3.5-14) and the 2013 NK054-REP-01210-00098 (Reference 3.5-15). SPT N-values are converted to  $N_{60}$  values (N value at 60 percent hammer energy) based on measured or assumed hammer energies for the automatic hammer and drill rigs used in the investigation, per the 2012 NK054-REF-01210-0418696 (Reference 3.5-14).

The drained friction angles for the soil layers are estimated using correlations based on relative density,  $N_{60}$ , and vertical effective stress for cohesionless soils provided in the 1986 DM 7.01 (Reference 3.5-16), the 1990 EPRI EL-6800 (Reference 3.5-17) and the 2016 Soil Properties and their Correlations (Reference 3.5-18). The different correlations are equally weighted to determine the final average drained friction angle value. The values for the coefficient of earth pressure at rest ( $K_0$ ) are determined using effective angle of friction ( $\phi_s$ ) and over-consolidation ratio based on the 2021 NEDO-33914 (Reference 3.5-10).

Bounding properties of the engineered fill are developed based on the information obtained from compaction tests that were completed for the upper till, intermediate glacio-lacustrine, and lower till units presented in the 2009 DNNP Existing Environmental Conditions NK054-REP-07730-00005 (Reference 3.5-12). Based on the result from standard compaction tests, the relative density ( $D_r$ ) and  $N_{60}$  values of the compacted soils are estimated. Relative density is estimated using the empirical relationship between  $D_r$  and compaction in the 2009 NK054-REP-07730-00005 (Reference 3.5-12). A relative compaction range of 85 to 100 percent is considered reasonable to cover the potential variations in placement and compaction of the on-site soils. The  $E_{st}$  of the compacted fill is determined from the estimated  $N_{60}$  values described in the 2016 Soil Properties and their Correlations (Reference 3.5-18) similar to the in-situ soils. The drained

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

friction angle for the engineered or compacted fill is assumed to be similar to the in-situ soils that will be excavated.

Bounding values for the linearized  $E_{st}$  of the rock masses at the DNNP site are estimated based on the intact rock modulus ( $E_{ri}$ ) and the rock mass classification determined from results of the site investigation program and an estimated Geologic Strength Index for the different bedrock formations. Results of Uniaxial Compression Tests performed on intact rock specimens and  $V_s$  and  $V_p$  measurements can serve as the basis for development of  $E_{ri}$  values. The  $\nu_{st}$  values for rock masses are developed based on  $V_s$  and  $V_p$  measurements and the level of rock fracturing.

The intact rock elastic properties are estimated from shear wave velocities using elastic theory as outlined in the 2021 NEDO-33914 (Reference 3.5-10). Results of laboratory measurements on recovered rock provided in the 2012 NK054-REF-01210-0418696 (Reference 3.5-14) and the 2013 NK054-REP-01210-00098 (Reference 3.5-15) are also used to estimate the intact rock elastic properties of the Blue Mountain (Whitby) and Lindsay Formations. The laboratory measured elastic modulus values in the Blue Mountain (Whitby) and Lindsay Formations were, on average, 94 and 75 percent, respectively, of the estimated values from the  $V_s$ . This comparison likely represents the different strain levels as well as potential damage from rock coring. Based on this comparison, the estimates of the modulus for intact rock from bedrock units below Lindsay Formation are reduced by a factor of 0.75. In the Blue Mountain (Whitby) and Lindsay Formations (Lindsay 1), the lower intact rock deformation modulus from the laboratory testing results is used.

The rock  $\nu_{st}$  values are based on the laboratory measured values and the estimates from  $V_s$  and  $V_p$  measurements. Based on this comparison the seismic wave estimated values are used without modification. Blue Mountain (Whitby) Formation is assigned  $\nu_{st}$  value of 0.58 based on an at-rest stress ratio ( $K_0$ ) that includes the estimated horizontal rock stresses at the site provided by Lo and Lukajic in (Reference 3.5-19) that are higher than the vertical stresses.

Table 3.5-1 provides a summary of bounding linearized static properties for in-situ soil and engineered fill layers in the as-built profiles. The summary of bounding static properties for the rock layers at the DNNP site are provided in Table 3.5-2.

UB values for soil effective unit weight and Poisson ratio are used as input for the static 1-g SSI analysis to conservatively address uncertainties in the consideration of earth pressure loads. In accordance with the guidance of NEDO 33914, Section 5.2.1.1, the soil Poisson ratios ( $\nu_{st}$ ) are calculated as follows using the at-rest lateral ( $k_0$ ) coefficient values provided in Table 3.5-1:

$$\nu_{st} = \frac{K_0}{1 + K_0}$$

LB soil and rock stiffness properties are used for the static analyses including the 1-g SSI analyses resulting in larger deformation at soil-structure interfaces and conservative design stress demands. Thermal stress analyses are performed using UB soil and rock stiffness properties resulting in conservative thermal stress demands.

### **3.5.2.2.2 Soil Bearing Stability**

The stability of soil supporting the BWRX-300 structural foundations is demonstrated in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2 and per the regulatory guidance of US NUREG-0800 (Reference 3.5-20), SRP 2.5.4.10, and IAEA Safety Guide No. NS-G-3.6 (Reference 3.5-21).

The bearing capacity of the rock supporting the RB mat foundation is discussed in Chapter 2, Subsection 2.7.3.3.

Since the RB is deeply embedded, the bearing surface of the common foundation is below the depth of frost action to meet the requirements of NBC (Reference 3.5-22), Article 4.2.4.4.

Chapter 2, Subsection 2.7.3.3 also discusses the bearing capacity of the component in-situ soil materials supporting the shallow foundations surrounding the RB.

The calculation of the dynamic bearing pressure demands under DBE loads from the results of the seismic SSI analyses is described in Subsection 3.3.1.2.

Per Article 4.35 of IAEA Safety Guide No. NS-G-3.6, safety factors against potential bearing capacity failure of the subsurface materials depend on the method of bearing capacity evaluation and site conditions. If a conventional bearing capacity method is used, safety factors are not less than 3 under static loads and 1.5 under loads that include DBE.

### **3.5.2.2.3 Foundation Stability**

Foundation stability is assessed against sliding and overturning due to earthquakes, wind and tornados, and flotation in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2, following the regulatory guidance of US NUREG-0800, SRP 3.8.5 and in accordance with Clause 5.9 of CSA N289.3.

Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB since, in accordance with Sections 7.2.1 and 7.2.2 of ASCE/SEI 43 (Reference 3.5-23), its centre of gravity is below the grade elevation, and the structure is inherently stable against sliding and overturning. The foundation stability of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay that are supported by surface mounted foundations is checked to ensure that there is no adverse interaction with the Seismic Category A RB during a DBE level event. Stability of the surface mounted foundations surrounding the RB under DBE loads is evaluated using the results of the seismic SSI analyses as described in Subsection 3.3.1.2.

Safety factors against sliding and overturning under normal operating conditions that include unfactored combination of dead loads, soil pressure loads, and design wind, and accidental conditions that include combination of dead loads, soil pressure loads, and DBE loads are presented in Table 3.5-3.

### **3.5.2.3 Loads and Load Combinations**

#### **3.5.2.3.1 Design Loads**

Design loads of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For foundation stability against flotation, the site-specific design basis flood is considered.

#### **3.5.2.3.2 Design Load Combinations**

Design load combinations of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For the stability against flotation of the integrated RB foundation, the load combination is in accordance with U.S. NUREG-0800, SRP 3.8.5, where the design basis flood is considered in combination with the dead load.

### **3.5.2.4 Design and Analysis Procedures**

The design of the deeply embedded foundation and foundation stability evaluations are in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1 and follow the BWRX-300 specific criteria and guidelines in NEDO-33914 Revision 2.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The containment and RB common mat foundation is analyzed using the methods where the transfer of loads from the foundation mat to the supporting foundation media is determined by elastic methods. Demands for the design of the common mat foundation are obtained from the structural analyses described in Subsection 3.5.1.1 performed on the integrated RB structural model that include the effects of interaction of the structure with the surrounding subgrade and the effects of the foundations of the surrounding Power Block buildings.

The common Steel Bricks™ foundation mat is represented by thick shell elements in the integrated FE model. Properties assigned to the shell elements representing the common Steel Bricks™ foundation in the dynamic FE model used for the seismic SSI analyses are described in Subsection 3.3.1.2. Properties assigned to the foundation shell elements in the integrated FE models used for the static and thermal stress analyses are described in Subsection 3.5.1.1.

The containment foundation is designed in accordance with NEDC-33926P, consistent with U.S. NRC RG 1.136 (Reference 3.5-24). The non-pressure retaining portion of the containment-RB common foundation mat is designed to ANSI/AISC N690, supplemented by U.S. NRC RG 1.243 and NEDC-33926P.

Effects of normal and differential settlement of BWRX-300 structures is considered in the design and include consideration of the effects of fluctuating ground water on the foundations per CNSC REGDOC-2.5.2, Section 7.15.1, and CSA N291, Clause 6.4.3.

As mentioned in Subsection 3.5.1.1, contact springs are used to represent the stiffness properties of the foundation-subgrade interface. Vertical spring force results obtained from these spring elements serve for calculations of foundation bearing stresses.

### **3.5.2.5 Foundation Design Criteria**

The structural acceptance criteria for the containment and RB common foundation are the same as those for their respective superstructures. Refer to Subsection 3.5.2.2 for safety factors considered for soil bearing and foundations stability.

### **3.5.2.6 Materials, Quality Control and Special Construction Techniques**

#### **3.5.2.6.1 Foundation Materials**

Materials used for the construction of the containment and RB common foundation mat are the same as those of the superstructures discussed in Subsections 3.5.3.5 and 3.5.5.5.

#### **3.5.2.6.2 Foundation Quality Control**

Refer to Subsections 3.5.3.5 and 3.5.5.5 for discussion.

#### **3.5.2.6.3 Foundation Special Construction Techniques**

Refer to NEDO-33914 Revision 2, Section 1.4 for the preferred construction approach for the deeply embedded RB.

### **3.5.2.7 Testing and In-Service Inspection Requirements**

The foundation inspection and testing follow the guidance of NEDO-33914 Revision 2, Sections 3.2.1 and 3.4, and also NEDC-33926P.

## **3.5.3 Containment**

The BWRX-300 containment comprises a Steel-plate Composite Containment Vessel (SCCV), a steel containment closure head and other Class MC components. As described in Subsection 3.5.1, the BWRX-300 SCCV is constructed of Steel Bricks™.

### **3.5.3.1 Applicable Codes, Standards and Other Specifications**

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 containment are listed in Chapter 1, Appendix B.

The design of the BWRX-300 containment boundary structures, including the SCCV, containment closure head and other Class MC components complies with the regulatory requirements in CNSC REGDOC-2.5.2. The analysis and design, fabrication and testing of the SCCV is in accordance with the provisions of NEDC-33926P, which are based on analytical and engineering principles, including use of experimental results. Additional analysis and design requirements in U.S. NUREG-0800, SRP 3.8.1 and U.S. NRC RG 1.136 for concrete containment are also met, as applicable. The compliance with the provisions of NEDC-33926P and the regulatory guidance of U.S. NUREG-0800, SRP 3.8.1 and U.S. NRC RG 1.136 ensures a level of safety and performance for the SCCV compliant with CNSC REGDOC-2.5.2.

The containment closure head, and the other Class MC components that are part of the containment pressure boundary are analyzed, designed and inspected following the provisions of ASME Section III, Division 1, Subsection NE, ensuring compliance with the regulatory guidance of CNSC REGDOC-2.5.2.

#### **3.5.3.1.1 Containment code Jurisdictional Boundary**

For code applicability, the SCCV is designed in accordance with ASME BPVC Section III requirements. The code jurisdictional boundary for application of Section III of ASME BPVC to the SCCV is shown in Figure 3.5-1. The SCCV boundary extends to the:

1. Outside diameter of the SCCV wall from mat foundation to containment top slab including the welds connecting the SCCV with the RB structural members
2. Portion of the foundation mat foundation under SCCV including the welds connecting the SCCV portion of the mat foundation with the remaining part of the RB mat foundation
3. Containment top slab from containment closure head opening to the outside diameter of the SCCV including the welds connecting the slab with the RB structural members

The BWRX-300 containment closure head and other containment boundary metal components are ASME Code Class MC. The code jurisdictional boundary for application of ASME BPVC Section III, Division 1, Subsection NE, Class MC to the containment closure head, access hatches and penetrations are shown in Figure 3.5-2, Figure 3.5-3 and Figure 3.5-4, respectively.

The SCCV along with the containment closure head, access hatches and penetrations, provide the primary containment function as a leak-tight pressure boundary confining radioactive substances in different plant conditions. Although the internal RPV support pedestal, bioshield and other containment internal structures are completely within the containment, these internal structures do not serve any pressure retaining function and are, thus, outside the scope of ASME Code applicability. The design of welds connecting the containment internal structures to the containment pressure boundary are under ASME jurisdiction. The connections of the RB walls and floors to the outside face of the SCCV wall are outside ASME code jurisdiction, with the exception of attachment welds. Attachment welds are designed to follow ASME quality assurance and welding procedures and inspection requirements.

### **3.5.3.2 Load and Load Combinations**

#### **3.5.3.2.1 Containment Design Loads**

Loads used in the design of the BWRX-300 containment structures, comprised of the SCCV, containment closure head, and other Class MC components, satisfy the loading requirements of

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

the applicable regulations, design codes and standards in Subsection 3.5.3.1. These loads are in accordance with the provisions of ASME III Division 1, Subsection NE, ASME III Division 2 (Reference 3.5-25) and NEDC-33926P.

Loads considered in the design of the BWRX-300 containment structures are:

- Normal Loads:
  - Dead load (D) which includes permanent dead weight of structural and shielding elements, permanently located equipment and hydrostatic pressure of liquids in various pools
  - Live loads (L,  $L_o$ ) which include any moveable equipment loads and other loads that vary in intensity and occurrence
  - Indirect Snow (S) and Rain (R) Loads
  - Thermal ( $T_o$ ) effects and loads during normal operating, startup, or shutdown conditions
  - Pressure ( $P_o$ ) loads resulting from the pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations
  - Pipe reactions ( $R_o$ ) during normal operating or shutdown conditions based on the most critical transient or steady-state conditions
  - Construction loads applied to the containment from start to completion of construction. The definitions for D, L and  $T_o$  given above are applicable, but are based on actual construction methods and/or conditions
  - Pressure Variant loads ( $P_v$ ) which are the external pressure loads arising from variation either inside or outside the SCCV
    - Indirect Lateral Soil and groundwater pressure loads (H)
- Pre-operational Testing Loads:
  - Thermal ( $T_t$ ) effects and loads during the SIT or Integrated Leak Rate Test (ILRT)
  - Test Pressure ( $P_t$ ) Loads applied during the SIT or ILRT
- Severe Environment Loads:
  - Indirect design Wind Load (W) defined in Subsection 3.3.2
- Extreme Environmental Loads:
  - Indirect Tornado ( $W_t$ ) Loads defined in Subsection 3.3.2
  - DBE seismic ( $E_s$ ) loads determined for DNNP site-specific conditions taking into account SSI effects, as discussed in Subsection 3.3.1, and include associated hydrodynamic loads and dynamic incremental soil pressures
- Abnormal Plant Loads:
  - Accidental Thermal effects ( $T_a$ ) due to LOCA
  - Accidental Pressure ( $P_a$ ) loads within the containment generated by a LOCA
  - Accidental Pipe ( $R_a$ ) reaction loads that consist of pipe reactions (including  $R_o$ ) from thermal conditions generated by design basis accidents such as LOCA and DBE

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Local effects on containment due to LOCA ( $R_r$ ) and Blast Loads ( $R_b$ ) which includes:
  - $R_{rr}$  load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA
  - $R_{rj}$  Load on the containment generated by the jet impingement from a ruptured high-energy pipe during the postulated event of the DBA
  - $R_{rm}$  load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA
  - Additional blast loads that may result from a postulated instantaneous break of a large pipe that could occur prior to the jet loads and that do need to be combined with the other loads
- Internal flooding loads resulting from a DBA
- Hard objects drop impact loadings, as applicable

Loads associated with DEC representing a subset of beyond design basis accident conditions are discussed in Subsection 3.5.6.

#### **3.5.3.2.2 Design Load Combinations for the SCCV**

The SCCV portion of the BWRX-300 containment is designed for load combinations and associated load factors for applicable loading conditions in accordance with NEDC-33926P, supplemented by U.S. NRC RG 1.136.

#### **3.5.3.2.3 Design Load Combinations for the Containment Closure Head and Other Class MC Components**

Load combinations and associated load factors used in the design of the containment closure head and other Class MC components are in compliance with U.S. NRC RG 1.57 (Reference 3.5-26) and U.S. NUREG-0800, SRP 3.8.2.

The portion of the BWRX-300 containment closure head and other Class MC components backed by concrete are designed for the load combinations and associated load factors in accordance with NEDC-33926P, supplemented by US NRC RG 1.136.

#### **3.5.3.3 Design and Analysis Procedures**

##### **3.5.3.3.1 Containment Structural Analysis Procedures**

As mentioned in Subsection 3.5.1.1, the BWRX-300 RB, including the containment, the containment internal structures and their common foundation, are analyzed as one integrated structure.

The connections between the SCCV and the RB members in the integrated FE model are modeled to reflect the appropriate load transfer for gravity, lateral and thermal loads.

Analysis procedures for the integrated structure are discussed in Subsection 3.5.1.1.

##### **3.5.3.3.2 Structural Design Method for SCCV**

The design of the SCCV structure conforms to the requirements of NEDC-33926P and meets the acceptance criteria discussed in Subsection 3.5.3.4.

Membrane forces, shear forces and bending moments used in the design of SCCV sections are obtained from the linear elastic computer analyses for the integrated RB and SCCV FE model discussed in Subsection 3.5.1.1. Subsection 3.5.5.3.2 provides further details for the critical section identification and design.

### **3.5.3.3.3 *Structural Design Methods for Containment Closure Head and Other Class MC Components***

The design procedures for the containment closure head and other Class MC components are as shown in Figure 3.5-5 and Figure 3.5-6, respectively.

The BWRX-300 containment closure head and other Class MC components are designed in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticles NE-3100 (General Design), NE-3200 (Design by Analysis), and NE-3300 (Design by Formula), as applicable. The design meets the acceptance criteria discussed in Subsection 3.5.3.4, including buckling and fatigue evaluations as required. The design by analysis utilizes the demands from the analyses of appropriate finite element models as described in Subsection 3.5.1.1. The stresses, including discontinuity stresses induced by the combination of applicable loads during different plant conditions, are evaluated, as applicable.

The access hatch cover with the bolted flange is designed in accordance with Subarticle NE-3326 of ASME BPVC, Section III, Division 1, Subsection NE.

### **3.5.3.4 *Structural Acceptance Criteria***

#### **3.5.3.4.1 *Design Basis Acceptance Criteria for SCCV***

The acceptance criteria for the design of the SCCV are in accordance with NEDC-33926P. The allowable stresses and strains in NEDC-33926P, for service and factored loads used in the design of the SCCV are provided in Table 3.5-4.

#### **3.5.3.4.2 *Design Basis Acceptance Criteria for Containment closure Head and Other Class MC Components***

The acceptance criteria for the design basis loads of the steel containment closure head and other MC components are the allowable stress limits specified in ASME BPVC, Section III, Division 1, Subsection NE-3220. The structural acceptance criteria for the Post-flooding condition, which is only applicable for other Class MC components excluding the containment closure head, is in accordance with U.S. NUREG-0800, SRP 3.8.2. Table 3.5-5 and Table 3.5-6 summarize the acceptance criteria for testing, design, Level A, C and D, and Post-flooding conditions, as applicable, for the containment closure head and other Class MC components, respectively. Stability against compression buckling is assured by an adequate factor of safety.

#### **3.5.3.4.3 *Containment Seismic Design Criteria***

The Seismic design criteria for the BWRX-300 containment are summarized in Table 3.3-1.

The seismic design of the BWRX-300 containment considers LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response without any significant permanent deformation when subjected to DBE, and complying with the regulatory requirements in CNSC REGDOC-2.5.2, Section 8.6.2.

Per CSA N289.3, Clause 7.5, the seismic design of the:

- SCCV is in accordance with NEDC-33926P
- Steel components at the containment boundary not backed by SCCV is in accordance with provisions of ASME BPVC, Section III, Division 1, Subsection NE

Also, in compliance with CNSC REGDOC-2.5.2, Section 8.6.2, the BWRX-300 containment meets the deformation acceptance criteria of ASCE/SEI 43, Section 5.2.3 and possesses ductility and energy absorbing capacity which permits inelastic deformation without failure under DEC's.



#### **3.5.3.4.4 Containment Design Criteria for Impulsive and Impactive Loads**

The BWRX-300 containment is designed for impulsive and impactive loads in compliance with requirements of Sections 7.15.1 and 7.15.3 of CNSC REGDOC-2.5.2 and the regulatory guidelines of U.S. NUREG-0800, SRP 3.8.1, Appendix A.

The design of the SCCV for impulsive and impactive loads follows the applicable requirements of the SCCV NEDC-33926P.

The design of the steel components of the containment not backed by SCCV follows the relevant regulatory guidance of U.S. NRC RG 1.57 and provisions of ASME BPVC, Section III, Division 1, Subsection NE.

#### **3.5.3.4.5 Containment Robustness Acceptance Criteria**

Complying with CNSC REGDOC-2.5.2, Section 6.1, the Level Four D-in-D described in Subsection 3.1.6 requires that the containment design be robust to provide adequate protection for the confinement function, including the use of complementary design features to prevent accident progression and to mitigate the consequences of DEC and BDBAs. Refer to Subsection 3.5.6.1 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 containment. These acceptance criteria satisfy the requirements in CNSC REGDOC-2.5.2, Sections 7.22.3 and 8.6.12, ensuring there is sufficient structural integrity to protect important systems in event of a design basis threat.

The leak tightness at the boundary of the containment structure, including the SCCV, containment closure head, and other Class MC components, under DEC internal pressure loads meets the requirements of CNSC REGDOC-2.5.2 and U.S. NRC RG 1.216 (Reference 3.5-27).

### **3.5.3.5 Materials, Quality Control and Special Construction Techniques**

#### **3.5.3.5.1 Containment Materials**

Materials used in the construction of the SCCV portion of the containment structure are in accordance with NEDC-33926P and U.S. NRC RG 1.136.

Steel materials used in the fabrication of the containment closure head and other Class MC components are in accordance with ASME Section III Subsection NE, Article NE-2000.

Details of materials used in the construction of the containment structures are provided in Chapter 9B, Subsection 9B.2.1.4.

#### **3.5.3.5.2 Containment Quality Control**

Quality control procedures are established for the containment structure in the construction, fabrication and installation specifications and implemented during fabrication, construction, installation, and inspection. These specifications cover the fabrication, furnishing, and installation of each structural item and specifies the inspection and documentation requirements to ensure that the requirements of NEDC-33926P, Articles NE-4000 and NE-5000 of ASME Section III, Division 1, Subsection NE, U.S. NRC RG 1.28 (Reference 3.5-28), U.S. NRC RG 1.136, and U.S. NUREG-0800, SRP 3.8.2 are met.

#### **3.5.3.5.3 Containment Special Construction Techniques**

The integrated RB, SCCV, RPV pedestal, and other structural components are constructed using modular construction technique as described in Subsection 3.5.5.5.

#### **3.5.3.6 Testing and In-Service Inspection Requirements**

Concrete and concrete constituents in the Steel Bricks™ modules of the SCCV are examined and tested in accordance with NEDC-33926P, as supplemented by the concrete sampling

requirements in NEDO-33914 Revision 2. Inspection of Steel Bricks™ welds is in accordance with NEDC-33926P.

#### **3.5.3.6.1 Structural Integrity Test (SIT)/PRE-Operational Proof Test**

The SCCV pre-service SIT plan and instrumentation is in compliance with NEDC-33926P and U.S. NRC RG 1.216. The SIT ensures compliance with containment pressure structure capability requirement for pressure tests in CNSC REGDOC-2.5.2, Section 8.6.3.

In accordance with NEDC-33926P, deformation, stress and strain measurements are made to evaluate the behavior of the containment and confirm that the actual structural response is within the limits predicted by analysis.

#### **3.5.3.6.2 Containment Pre-Service and In-Service Inspection**

The SCCV pre-service and periodic in-service inspection plan is in accordance with NEDC-33926P to comply with the requirements of CNSC REGDOC-2.5.2.

#### **3.5.3.6.3 Integrated Leak Rate Testing**

The SCCV is designed such that the periodic ILRT can be conducted at the design pressure to demonstrate the leak tightness integrity of the containment boundary in compliance with Section 8.6.4 of CNSC REGDOC-2.5.2. The ILRT is performed per criteria outlined in Chapter 6, Subsection 6.3.7.

The flange seals of the containment closure head and Class MC components that have potential for significant contribution to leakage are designed to be individually testable. Where resilient seals such as elastomeric seals are used, they have the capability for performing leak testing at the containment design pressure in compliance with Section 8.6.5 of CNSC REGDOC-2.5.2.

### **3.5.4 Containment Internal Structures**

The BWRX-300 containment internal structures comprise the Steel Bricks™ RPV pedestal, the steel-plate composite bioshield surrounding the RPV pedestal and structural steel Containment Equipment and Piping Support Structure (CEPSS), including the support floor at Level -8.5 m, and support floors at Level -21 m and -29 m.

#### **3.5.4.1 Applicable Codes, Standards and Other Specifications**

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 containment internal structures are listed in Chapter 1, Appendix B.

Similar to RB, the analysis and design, fabrication and testing of the containment internal structures is in accordance with the ANSI/AISC N690, including the supplemental requirements in U.S. NRC RG 1.243 and NEDC-33926P. This methodology ensures a level of safety and performance for the containment internal structures commensurate to that required by CSA N291 and ensures compliance with CNSC REGDOC-2.5.2.

Refer to Figure 3.5-1 for the jurisdictional boundary for the RPV pedestal, the bioshield and internal structural steel.

#### **3.5.4.2 Loads and Load Combinations**

Since the containment internal structures are completely contained within and are integrated with the RB and SCCV, the design of containment internal structures considers both design loads applied directly to the containment internal structures and those applied indirectly through the RB and SCCV.

#### **3.5.4.2.1 Design Loads**

Refer to Subsections 3.5.3.2 and 3.5.5.2 for the description of design loads applicable for the SCCV and RB structures that are also generally applicable for the design of containment internal structures. Since containment internal structures are inside the containment, some of the design loads applicable for the RB are not directly applicable for the containment internal structures. Additionally, the internal flooding condition associated with post-accident flooding is not considered in accordance with U.S. NUREG-0800, SRP 3.8.1 as noted in Table 9B-1 in Chapter 9B.

The design loads also include the reactions from the RPV at the support locations on the containment internal structures and other bracket and attachment loads applicable during different plant conditions. The RPV lumped mass beam model representing the mass and stiffness properties of the RPV is included in the integrated FE model discussed in Subsection 3.3.1.2, and the dead load and seismic load reactions from the RPV are obtained directly from the static and seismic analyses. Other normal and accidental plant operating loads are applied to the model as reaction force loads.

#### **3.5.4.2.2 Design Load Combinations**

Load combinations and load factors for the design of the Steel Bricks™ structures and structural steel that form the containment internal structures are in accordance with ANSI/AISC N690, including the supplemental regulatory guidance of U.S. NRC RG 1.243.

#### **3.5.4.3 Design and Analysis Procedures**

##### **3.5.4.3.1 Structural Analysis Procedures**

Analysis procedures for the containment internal structures are the same as those for the integrated RB structure discussed in Subsection 3.5.1.1 since containment internal structures are included in the integrated FE model used in the analyses.

The connections between the containment internal steel structures and the RPV, RPV pedestal, bioshield and SCCV are appropriately modeled in the integrated FE model to reflect the appropriate load transfer for gravity and lateral loads.

Local models may be used, if needed, for detailed design at opening and connection locations.

##### **3.5.4.3.2 Structural Design Methods**

For the design of containment internal structures, the design methodology is the same as that used for the design of the RB structure, discussed in Subsection 3.5.5.3.

#### **3.5.4.4 Structural Acceptance Criteria**

##### **3.5.4.4.1 Design Basis Acceptance Criteria**

The design basis acceptance criteria of the containment internal structures, including the Steel Bricks™ RPV pedestal, the steel-plate composite bioshield and containment internal steel structures, are same as those for the corresponding RB structural components described in Subsection 3.5.5.4.

##### **3.5.4.4.2 Robustness Acceptance Criteria**

The methodology and acceptance criteria for the robustness of the containment internal structures are described in Subsection 3.5.6.1.

### **3.5.4.5 Materials, Quality Control and Special Construction Techniques**

#### **3.5.4.5.1 Materials**

The concrete and structural steel materials used for the construction of containment internal structures are same as those for the RB structure as described in Subsection 3.5.5.5, except that pool liners are not applicable.

#### **3.5.4.5.2 Quality Control**

The quality control requirements for containment internal structures are same as those for the RB structure as described in Subsection 3.5.5.5.

#### **3.5.4.5.3 Special Construction Techniques**

The integrated RB, SCCV, RPV pedestal, and other structural components are constructed using modular construction technique as described in Subsection 3.5.5.5.

#### **3.5.4.6 Testing and In-Service Inspection Requirements**

A formal program of testing and in-service inspection is not required for containment internal structures since they are not directly related to the functioning of the containment system. However, during the operating life of the plant, the condition of the containment internal structures is monitored per 10 CFR 50.65 in accordance with U.S. NRC RG 1.160 (Reference 3.5-29).

### **3.5.5 Reactor Building**

#### **3.5.5.1 Applicable Codes, Standards and Other Specifications**

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 RB are listed in Chapter 1, Appendix B.

Specifically, the analysis and design, fabrication and testing of the RB structure (including the Steel Bricks™ walls, slabs and mat foundation and the structural steel components, see Figure 3.5-1) is in accordance with the ANSI/AISC N690, including the supplemental requirements in U.S. NRC RG 1.243 and NEDC-33926P. This methodology ensures a level of safety and performance for the RB commensurate to that required by CSA N291 and ensures compliance with CNSC REGDOC-2.5.2.

The RB polar crane is designed and constructed to meet the requirements of ASME NOG-1 (Reference 3.5-30).

Crane loading is developed in accordance with NBC and ASCE/SEI 7 (Reference 3.5-31), Section 4.9.

#### **3.5.5.2 Loads and Load Combinations**

In addition to the loads applicable directly to the RB, loads considered in the design of the RB include loads applied to the SCCV that have an effect on the RB structure due to the common mat foundation, floor slabs, RB shear walls and other integrating structural components.

##### **3.5.5.2.1 Design Loads**

The RB structure is analyzed and designed in accordance with ANSI/AISC N690 for design basis load cases in compliance with CSA N291.

Loads, such as accident pressure and thermal transient loads due to a LOCA, internal to SCCV are considered for the design of structural components of the RB that are integrated with the SCCV.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

RB design loads consist of:

- Service category of loads that occur during construction, pre-operational testing, or normal operation. They include:
  - Dead loads (D) which consist of the weight of structures, weight of permanently attached major equipment, tanks, machinery, and cranes; weight of piping, cable, cable trays, duct supports; and hydrostatic pressure of liquids in various pools
  - Live loads ( $L$ ,  $L_r$ ) which consist of floor area loads, laydown loads, nuclear fuel, and equipment handling loads
  - Lateral Soil and groundwater pressure loads (H)
  - Snow/rain loads (S/R) discussed in Subsection 3.3.2
  - Normal plant operation and pre-operation pressure testing loads which consist of operation service pressure loads, pre-operation proof test pressure load, normal thermal conditions ( $T_o$ ) and operation service pipe reaction loads ( $R_o$ )
  - Construction Loads
  - Settlement Loads
  - Crane Loads developed as discussed in Subsection 3.5.5.1.
- Abnormal and environmental category of loads that occur during postulated accident and/or severe or extreme environmental events. They include:
  - Abnormal plant operation loads which include accident pressure ( $P_a$ ) and thermal ( $T_a$ ) loads, accident pipe reaction loads ( $R_a$ ), missile generation, pipe whip ( $Y_r$ ), jet impingement from large pipe breaks ( $Y_j$ ), blast pressure ( $Y_m$ ), compartment pressurization and drop of large loads
  - Wind and Tornado loads ( $W$ ,  $W_t$ ) discussed in Subsection 3.3.2
  - Seismic loads ( $E_s$ ) discussed in Subsection 3.3.1, including hydrodynamic loads on the pool walls calculated based on the approach described in ASCE/SEI 4 (Reference 3.5-32) and ACI 350.3 (Reference 3.5-33), and dynamic incremental soil pressures
- Hard objects drop impact loadings, as applicable
- Design Basis Threat loads discussed in Subsection 3.3.7.4

Loads associated with DEC representing a subset of beyond design basis accident conditions are discussed in Subsection 3.5.6.

#### **3.5.5.2.2 Design Load Combinations**

Load combinations and load factors for the design of the Steel Bricks™ module structures and structural steel in the RB are in accordance with the provisions of ANSI/AISC N690, Chapter NB2.6 including the supplemental regulatory guidance of U.S. NRC RG 1.243, Regulatory Positions 2.1 and 2.2.

#### **3.5.5.3 Design and Analysis Procedures**

##### **3.5.5.3.1 Structural Analysis Procedures**

Refer to Subsection 3.5.1.1 for analysis procedures.

#### **3.5.5.3.2 Structural Design Methods**

The design of the RB structure conforms to the requirements of ANSI/AISC N690, including the regulatory guidance in U.S. NRC RG 1.243 and meets the acceptance criteria discussed in Subsection 3.5.5.4 to ensure a level of safety and performance commensurate with the requirements in CSA N291.

Membrane forces, shear forces and bending moments used in the design of the RB Steel Bricks™ and steel sections are obtained from the linear elastic computer analyses for the integrated RB FE model discussed in Subsection 3.5.1.1.

Results from the FE analyses are evaluated to identify critical cross-sections where maximum structural demands occur for different controlling loads and load combinations. Key responses reviewed include:

- Membrane forces for the SCCV,
- In-plane shear demands at the base of major walls and at rock-soil interface elevation,
- Vertical bending moments and out-of-plane shear demands on the RB outer shaft and SCCV walls, at base of walls and at intermediate floor elevations and
- Out-of-plane demands for major floor slabs and RB foundation mat at mid-span and support locations.

The structural demands at the critical locations are used to perform the design of the critical cross-sections and connections using the applicable codes of record.

#### **3.5.5.4 Structural Acceptance Criteria**

##### **3.5.5.4.1 Design Basis Acceptance Criteria**

The RB Steel Bricks™ module structures and structural steel, including welded and bolted connections, are designed to meet the acceptance criteria outlined in ANSI/AISC N690.

The RB structure is evaluated for serviceability considerations including deflection, vibration, permanent deformation, cracking, and settlement. Serviceability evaluations meet the acceptance criteria in ANSI/AISC N690, Chapter NL.

#### **Seismic Design Criteria**

The Seismic design criteria for the BWRX-300 RB are summarized in Table 3.3-1.

The seismic design of the RB structure considers LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response without any significant permanent deformations when subjected to DBE and complying with the regulatory requirements in CNSC REGDOC-2.5.2, Section 8.6.2.

The BWRX-300 RB structure meets the deformation acceptance criteria of ASCE/SEI 43, Section 5.2.3 and possesses ductility and energy absorbing capacity which permits inelastic deformations without failure under DEC.

#### **Evaluation Criteria for Structure Interaction Under Seismic and Extreme Wind**

The interaction of the RB structure with the adjacent RWB, CB, TB and Reactor Auxiliary Bay is discussed in Subsections 3.3.1.2 and 3.3.2.8.

The stability of foundations under DBE and design basis tornado wind loads are checked following the criteria in Subsection 3.5.2.2.

## **RB Design for Impulsive and Impactive Loads**

The RB structure is designed for impulsive and impactive loads per the requirements of Sections 7.15.1 and 7.15.3 of CNSC REGDOC-2.5.2 and the regulatory guidelines of U.S. NUREG-0800, SRP 3.8.4.

The RB design for impulsive and impactive loads follows the provisions of ANSI/AISC N690 and the relevant regulatory guidance of U.S. NRC RG 1.243.

Criteria used to define the heavy loads considered in the RB design are described in Subsection 3.4.5.1.

### **3.5.5.4.2 Robustness Acceptance Criteria for RB Structure**

Refer to Subsection 3.5.6.1 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 RB structure, which satisfy the requirements in CNSC REGDOC-2.5.2, Section 7.22.3.

## **3.5.5.5 Materials, Quality Control and Special Construction Techniques**

### **3.5.5.5.1 Materials**

Materials used in construction of the RB structure outside of the containment are in accordance with ANSI/AISC N690, Section NA3.

Details of materials used in the construction of the RB are provided in Chapter 9B, Subsection 9B.2.3.4.

### **3.5.5.5.2 Quality Control**

Quality control procedures are established and implemented during the construction and inspection phases of the RB structure. These procedures cover the fabrication, furnishing, and installation of each structural item in the RB and specify the inspection and documentation requirements in accordance with the requirements in ANSI/AISC N690, Section NA5, Chapter NN with supplemental guidance provided in U.S. NRC RG 1.243.

### **3.5.5.5.3 Special Construction Techniques**

The BWRX-300 Seismic Category A structures at the DNNP site are built using a modular construction technique using Steel Bricks™. (see Section 3.5.1).

The quality control procedures used in the structural modularization process implemented in the construction of the Steel Bricks are outlined in Subsection 3.5.5.5.2. These procedures are employed at the fabrication shop and the construction-site (both outside and inside the deep excavation pit necessary for the construction of RB), including pre-fabrication and pre-assembly, to ensure the Steel Bricks™ modular assemblies meet the necessary material quality, fabrication, and installation requirements per the applicable code of records.

For the preferred method of construction for the deeply embedded BWRX-300 RB shaft, refer to Section 1.4 of NEDO-33914 Revision 2.

For plant construction and commissioning activities, refer to Chapter 14.

### **3.5.5.6 Testing and In-Service Inspection Requirements**

Per CNSC REGDOC-2.5.2, Section 7.15.2, periodic inspection, and in-service monitoring programs are implemented to ensure the RB structure continues to meet its functional and performance requirements.

Sections 3.2 through 3.4 of NEDO-33914 Revision 2 describe the approaches and guidelines for the BWRX-300 in-service testing, monitoring, and monitoring programs.

NEDC-33926P describes the in-service inspection and testing guidelines for the Steel Bricks™ to ensure that the integrated RB structures satisfy their functional and performance requirements through all phases of the plant's life cycle. The BWRX-300 implements a Structures Monitoring and Aging Management Program (SMAMP) that monitors the condition of structures and manages aging effects in accordance with CSA N291, clauses 9 and 10 and in compliance with CNSC REGDOC-2.5.2, Section 7.17. The program demonstrates that the facility is constructed to the requirements in the design drawings and specifications. A research and development program is also established to demonstrate the adequacy of Steel Bricks™ to maintain the structural integrity of the integrated RB structures and of inspection methods used in compliance with CNSC REGDOC-2.5.2, Section 5.4.

### **3.5.6 Robustness Design of Seismic Category A Structures**

Consistent with the Level Four D-in-D requirements discussed in Subsection 3.1.6 and in Section 6.1 of CNSC REGDOC-2.5.2, the BWRX-300 containment and RB are robust structures, tolerant of a large spectrum of faults with a gradual degradation in their effectiveness, that would not fail catastrophically under operational states, DBAs and DEC.

Evaluations performed to establish an understanding of safety margins, or the robustness of the design are consistent with the regulatory guidance of CNSC REGDOC-2.4.1 (Reference 3.5-34), Section 4.2.3 and U.S. NUREG-0800, SRP 19.0.

#### **3.5.6.1 Design Extension Conditions**

In accordance with Section 7.15.1 of CNSC REGDOC-2.5.2, DEC considered in the design of the BWRX-300 Seismic Category A structures include severe accident conditions due to both internal and external hazards, whose probability of occurrence is lower than the probability of occurrence of the DBA.

Loads, load combinations, strength and safety requirements for assessing the BWRX-300 Seismic Category A structures (i.e., the integrated RB) are defined in accordance with Clause 6.1.4 of CSA N291.

Consistent with Section 7.3.4 of CNSC REGDOC-2.5.2 and Clause 5.6 of CSA N290.16 (Reference 3.5-35), deterministic safety analyses are used to determine the applicable DEC and evaluate the consequences of the DEC.

In accordance with the guidelines of CSA N290.16, Clause 4.3.5, a best estimate approach is used to obtain a reasonable confidence in the assessed response to DEC.

A reasonable level of survivability of the structure under postulated DEC is demonstrated following requirements of Clause 6.1.3.1 of CSA N290.16. Per Clause 4.5 of CSA N290.16, less stringent assumptions than those applied for design basis, such as the permissible variances in Annex C of CSA N290.16, may be used when evaluating SSC performance under DEC.

##### **3.5.6.1.1 Containment Severe Design Extension Condition Evaluations**

Complying with Section 8.6.12 of CNSC REGDOC-2.5.2, the BWRX-300 containment design ensures the ability of the containment system to withstand loads associated with DEC.

Consistent with CNSC REGDOC-2.5.2, Section 8.6.2, the containment structure is designed to possess ductility and energy absorbing capacity, which permits inelastic deformation without failure under DEC.

The beyond design basis evaluations of the containment ensure the structural integrity and leak tightness of the containment structure under all applicable DEC loading cases in compliance with the regulatory guidance of CNSC REGDOC-2.5.2.



### **Containment Ultimate Pressure Capacity**

The ultimate internal pressure capacity of the containment structure, including the SCCV, containment closure head and penetrations, is determined to ensure its structural integrity and leak tightness under DEC internal pressure loads to meet the requirements in CNSC REGDOC-2.5.2, Section 7.15.1, U.S. NRC RG 1.216, and U.S. NUREG-0800, SRP 3.8.1.

This ultimate pressure capacity is obtained from the results of non-linear finite element analysis consistent with the guidelines of Regulatory Position 1 of U.S. NRC RG 1.216.

### **Robustness Against Combustible Gas Pressure Loads**

The BWRX-300 design demonstrates the ability of the containment to withstand DEC loads associated with combustion of gases consistent with requirements of Section 8.6.12 of CNSC REGDOC-2.5.2.

The containment is designed to ensure that its structural integrity is maintained to sustain the combustible gas pressure loads applicable for BWRX-300 consistent with the requirements in U.S. NRC RG. 1.136 and U.S. NRC RG 1.57.

### **Containment Severe Accident Performance Goal**

Consistent with guidance in CNSC REGDOC-2.5.2, Section 8.6.12, the BWRX-300 design is a fail-safe design that ensures that under DEC conditions with core damage, the containment:

- A. Maintains its role as a reliable leak-tight barrier for a minimum of 24 hrs following the onset of core damage
- B. Continues to provide a barrier against the uncontrolled release of fission products following the initial 24 hrs period

The methodology used to evaluate the robustness of the containment is per Regulatory Position 3 of U.S. NRC RG 1.216. The evaluation identifies pressure and temperature loadings associated with the more likely DEC challenges by considering the sequences of plant damage states that represent 90% or more of the core damage frequency. Analyses of global and local finite element models are performed to calculate the enveloping containment response for the identified accident challenges.

Criteria for factored load category in NEDC-33926P for the SCCV is used to demonstrate the containment deterministic performance goal for the initial 24 hours. The deterministic performance goal after the initial 24-hour period is demonstrated by showing that the containment leakage in a severe accident remains below the design leakage rate limit, consistent with CNSC REGDOC-2.5.2, Sections 8.6.4 and 8.6.12, for sufficient time to allow implementation of emergency measures.

During an extremely improbable severe accident in the BWRX-300, molten core debris may be present on the containment floor. A protective layer of refractory concrete prevents corium (as shown in Chapter 9B, Figure 9B-1) from degrading the SCCV inner steel faceplate that acts as the primary leak-tight boundary. Additional protection is provided by the outer steel faceplate for the SCCV foundation mat. The lower SCCV design has a provision for the installation of a severe accident core melt capture and retention structure with a spreadable area to prevent contact between the molten core and the containment liner and concrete. Refer to Chapter 15, Appendix 15B for more details on this corium shield and other complementary design features for BDBAs.

#### **3.5.6.1.2 Beyond Design Basis Seismic Robustness**

In accordance with CNSC REGDOC-2.5.2, Section 7.13.1, the design of the BWRX-300 Seismic Category A and Seismic Category B SSC credited to function during and after a Beyond-Design

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Basis Earthquake (BDBE) ensures their capability to maintain their structural integrity and to perform their intended safety function.

The BDBE is defined to meet the DEC identification requirements of CNSC REGDOC-2.5.2, Section 7.3.4. Per CNSC REGDOC-2.5.2, Section 7.13.1, a High Confidence ( $\geq 95\%$ ) of Low Probability ( $\leq 5\%$ ) of Failure (HCLPF) of at least 1.67 times that for the DBE is demonstrated for the SSC credited to function during and after a BDBE.

The methodology in Electrical Power Research Institute (EPRI) TR-103959 (Reference 3.5-36), TR-1002988 (Reference 3.5-37) and TR-1019200 (Reference 3.5-38), consistent with the recommendations of TR- 3002012994 (Reference 3.5-39) is used for the evaluations of seismic fragilities of BWRX-300 Seismic Category A and B SSC.

Following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, to ensure adequate margins for the BDBE, the seismic design satisfies the ductility detailing and design requirements for steel and steel-plate composite structures of ANSI/AISC N690, with the supplementary guidance of U.S. NRC RG 1.243 and NEDC-33926P. This approach meets the intent of CSA S16 (Reference 3.5-40), for Seismic Category A steel structures members and connections.

### **Checking Level Earthquake**

Per Clause 5.4.5 of CSA N289.1 (Reference 3.5-41), a Checking Level Earthquake (CLE) defines the earthquake level for BDBE evaluations to ensure prescribed safety margins for earthquakes exceeding the DBE.

The BWRX-300 plant is assessed during the design process, in accordance with Clause 8.2 of CSA N289.3, using CLE to:

- Provide detailing for post-elastic behavior and energy absorption during BDBE events
- Identify any SSC that can have insufficient seismic ruggedness, ductility, or inelastic response capability to withstand and perform their safety function during and after BDBE
- To ensure no cliff-edge effects

The site-specific CLE ground motion spectra are defined as 1.5 times the DBE, which is at a level sufficiently larger than the DBE to support meeting the acceptable plant HCLPF criteria of CNSC REGDOC-2.5.2, Section 7.13.1. The site-specific CLE is representative of a seismic hazard exceedance probability that is lower than the seismic hazard probability of the DBE and meets the requirements of Clause C.3.3 of CSA N289.1.

The selected CLE maintains consistency with the performance objectives expressed in Chapter 1 of ASCE/SEI 43 and the precedence set for definition of BDBE motion in Chapter 9 of ASCE/SEI 43. The performance objectives in ASCE/SEI 43 aim to achieve 10% unacceptable performance for 150% of DBE level per U.S. NRC RG 1.208 (Reference 3.5-42). It is recognized that the redundancy in the SSC credited to function during and after a CLE is included in the calculation of a plant level HCLPF of at least 1.67 times the DBE.

CLE in-structure demands for BDBE evaluations are obtained from BE approach seismic response analyses performed following the guidance of CSA N289.1, Clause C.4.2, consistent with the criteria in Subsection 3.3.1.3. The SSI input soil profiles for the BDBE evaluations are obtained at strain levels consistent with the CLE motion. The SSI analyses for BDBE evaluations may use Response Level 3 damping values in accordance with ASCE/SEI 43

In accordance with Section 5.2.7 of CSA N289.1, CLE is considered in combination only with normal operating loads.

### 3.5.6.2 Design for Malevolent Acts

The BWRX-300 uses a security by design process that involves security reviews during plant design to resolve DBT and BDBT security issues at the earliest stage, when changes have the least impact on cost and performance. Placement and number of doors, wall thicknesses to optimize resistance to explosive breaching, and equipment placement to facilitate better target set diversity are all achievable when security is integrated at an early stage. Continual design reviews against the DBT and BDBT capabilities during the entire design evolution ensure that emergent issues are identified and addressed as early in the process as possible.

The defensive strategy approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter. By analyzing the potential adversary pathways to critical components, determining adversary resources required to execute the path, and slowing the adversary movements and depleting the adversaries' resources before the path can be completed to the extent possible, the design limits the ability of malicious individuals to cause damage to key systems. This, along with the inherent slower accident progression of the BWRX-300 reactor, reduces or eliminates the reliance on immediate on-site armed responders to prevent substantial off-site radiological releases, which allows for longer term off-site response, interdiction, and neutralization.

#### Malevolent Acts Design Methods

The BWRX-300 design for DBTs and BDBTs satisfies the requirements of CNSC REGDOC-2.5.2, Section 7.22.2.

The design considers the following two types of structural failure modes with distinct loading characteristics and structural responses:

1. Local effects that in general would not result in structural collapse but may affect the functions of safety class SSC
2. Global failure modes characterized by major structural damage, such as significant perforation or collapse of large portions of the building walls, floors, and load carrying frames

These failure modes are considered separately with a consideration given that for some threats, such as an aircraft crash, they may act simultaneously or quasi-simultaneously.

Applicable local damage modes are considered in the design and empirical formulas are used to assess the structural behavior under local and concentrated loading.

The BWRX-300 design applies the Nuclear Energy Institute's methodology in NEI 07-13 (Reference 3.5-43) for aircraft crash evaluations with CNSC input and other detailed computer analytical methods, where appropriate, to evaluate the consequences of regulatory defined threats on a BWRX-300 reactor site. The CNSC acceptance criteria are then applied to the results.

Evaluations include:

- RB structural integrity including enclosed safety features as applicable:
  - Global failure (plastic collapse)
  - Local perforation (hard missile)
  - The acceptance criteria for both local and global behavior are satisfied simultaneously
- Containment and fuel pool heat removal capability
- Reactivity control following regulatory defined threats

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- Containment isolation following regulatory defined threats
- Fuel intrusion prevention
- Shock and vibration impact of critical equipment
- Short and long-term mitigation efforts required following commercial aircraft impact

### **Malevolent Acts Design Acceptance Criteria**

The design of the BWRX-300 Seismic Category A structures meets the following acceptance criteria for local response under malevolent acts depending on the structural system used:

1. For DBTs, no scabbing of the rear face of structural elements, possibly with limited, easily repairable, superficial spalling of concrete
2. For severe BDBTs, no scabbing of the rear face of structural element, or possible limited scabbing if confined by the steel liner that should remain leak-tight
3. For extreme BDBTs, no perforation, according to the applicable formula with a corresponding increase factor of 1.2 applied to the calculated thickness
4. For Steel Bricks™ members, the steel faceplate thickness to prevent perforation is at least 1.25 times that required by use of rational methods in accordance with ANSI/AISC N690 and NEDC-33926P

The structural acceptance criteria for global response are related to:

- The limitation of structural deflections for DBT and severe BDBT; or
- Overall damage for extreme BDBT

Special attention is given to:

- Damage to the containment and internal structures due to extensive deformations of the containment
- Shock damage to fragile components directly attached to the containment wall
- Induced vibration
- Post-event fireball explosions or blast waves
- Structural integrity of the polar crane

The acceptance criteria for local and global structural response are satisfied simultaneously.

Design criteria for the BWRX-300 RB specifies no global failure, no perforation, no spalling, and no fuel intrusion from the regulatory defined threats.

The design of BWRX-300 containment meets the malevolent acts acceptance criteria in NEDC-33926P that is consistent with the regulatory guidance in Table 1 of CNSC REGDOC-2.5.2, Appendix A.

The BWRX-300 Security Annex describes design methods and acceptance criteria for malevolent acts in greater details.

### **3.5.7 References**

- 3.5-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.5-2 CNSC Regulatory Document REGDOC-1.1.5, "Reactor Facilities: Supplemental Information for Small Modular Reactor Proponents."
- 3.5-3 CNSC Regulatory Document REGDOC-3.5.3, "CNSC Processes and Practices, Regulatory Framework."
- 3.5-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.5-5 NEDC-33926P, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.5-6 ASME BPVC-III NE-2021, "BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components," American Society of Mechanical Engineers.
- 3.5-7 ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," American Institute of Steel Construction.
- 3.5-8 USNRC Regulatory Guide 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments."
- 3.5-9 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.5-10 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC. (Reference 2.7-35),
- 3.5-11 NK054-REP-01210-00011 R001, "Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation. 2009 (Reference 2.7-1)
- 3.5-12 NK054-REP-07730-00005 Rev. R000, Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation. 2009 (Reference 2.7-41)
- 3.5-13 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.5-14 NK054-REP-01210-0418696, "Geologic and Geophysical Evaluation, Darlington Site Investigation – Phase III (Field Work), AMEC Report No. D0053/RP/002 R01, Volumes 1 and 2," Ontario Power Generation. 2012 (Reference 2.7-36)
- 3.5-15 NK054-REP-01210-00098 R000, Geotechnical Data Report – R2, Darlington New Nuclear Project Geotechnical Investigation, EXP Services Inc. Project No. BRM-00025482-A0," Ontario Power Generation. 2013 (Reference 2.7-37).
- 3.5-16 DM 7.01, "Soil Mechanics," Naval Facilities Engineering Command. 1986 (Reference 2.7-38),
- 3.5-17 EPRI EL-6800, "Manual on Estimating Soil Properties for Foundation Design," Electric Power Research Institute. 1990 (Reference 2.7-39)
- 3.5-18 Carter, M. and Bentley, S., "Soil Properties and their Correlations," John Wiley & Sons, West Sussex, UK, 2016. (Reference 2.7-40)
- 3.5-19 Lo, K.Y., and B. Lukajic, "Predicted and Measured Stresses and Displacements around the Darlington Intake Tunnel," Canadian Geotechnical Journal, 21:147-165. (Reference 2.7-33)

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.5-20 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."
- 3.5-21 IAEA Safety Standards Series No. NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants," International Atomic Energy Agency.
- 3.5-22 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.5-23 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.5-24 USNRC Regulatory Guide 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, U.S. Nuclear Regulatory Commission."
- 3.5-25 ASME BPVC-III-2, "Section III: Rules for Construction of Nuclear Facility Components – Division 2- Code for Concrete Containments," American Society of Mechanical Engineers.
- 3.5-26 USNRC Regulatory Guide 1.57, "Design Limits and Load Combinations for Metal Primary Reactor Containment System Components."
- 3.5-27 USNRC Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure."
- 3.5-28 USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)."
- 3.5-29 USNRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
- 3.5-30 ASME NOG-1, "Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.
- 3.5-31 ASCE/SEI 7, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," American Society of Civil Engineers.
- 3.5-32 ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers.
- 3.5-33 ACI 350.3-20, "Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures (ACI 350.3-20) and Commentary," American Concrete Institute.
- 3.5-34 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 3.5-35 CSA N290.16, "Requirements for beyond design basis accidents," CSA Group.
- 3.5-36 EPRI TR-103959, "Methodology for Developing Seismic Fragilities," Electric Power Research Institute.
- 3.5-37 EPRI TR-1002988, "Seismic Fragility Application Guide," Electric Power Research Institute.
- 3.5-38 EPRI TR-1019200, "Seismic Fragility Application Guide Update", Electric Power Research Institute.
- 3.5-39 EPRI TR-3002012994, "Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments," Electric Power Research Institute.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.5-40 CSA S16, "Design of Steel Structures," CSA Group.
- 3.5-41 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group
- 3.5-42 USNRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."
- 3.5-43 NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Nuclear Energy Institute.
- 3.5-44 ASME BPVC-II D, "Section II: Materials-Part D-Properties (Customary), American Society of Mechanical Engineers.
- 3.5-45 ASME BPVC-III, "Appendix XXVII: Design by Analysis for Service Level D," American Society of Mechanical Engineers.

**Table 3.5-1: As-Built Static Properties for Soil Layers**

Layer	Layer Thickness (m)	Total Unit Weight (kN/m <sup>3</sup> )	Drained Friction Angle (degrees)		Elastic Modulus (MPa)		At-Rest Lateral Earth Pressure Coefficient	
		Ave.	Ave.	Range	Lower	Upper	Ave.	Range
Fill 1	1.0	22.0	34	29 – 37	15.1	60.8	0.55	0.51 – 0.63
Fill 2	1.0	22.0	34	29 – 37	17.0	77.5	0.55	0.51 – 0.63
Fill 3	1.0	22.0	34	29 – 37	18.8	91.3	0.55	0.51 – 0.63
Fill 4	1.0	22.0	34	29 – 37	20.5	104	0.55	0.51 – 0.63
Fill 5	1.0	22.0	34	29 – 37	22.4	116	0.55	0.51 – 0.63
Fill 6	1.0	22.0	34	29 – 37	24.0	127	0.55	0.51 – 0.63
Fill 7	2.0	22.0	34	29 – 37	25.8	138	0.55	0.51 – 0.63
Upper till	1.1	23.8	37	37	37.0	482	0.32	0.32 – 0.33
Interm. Glacio-lacustrine (Sandy)	7.2	20.9	36	36	36.2	411	0.35	0.34 – 0.35
Interm. Glacio-lacustrine (Silty)	2.8	21.1	30	28 – 32	33.9	379	0.83	0.80 – 0.86
Lower till	4.8	23.5	34	33 – 35	38.1	496	0.78	0.77 – 0.78



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.5-2: Summary of Static Rock Properties**

Layer	Total Unit Weight (kN/m <sup>3</sup> )	Intact Rock Deformation Modulus (GPa)	Rock Mass Deformation Modulus (GPa)		Poisson's Ratio
			Average	Range	
Blue Mountain (Whitby)	26.4	31.8	6.4	4.7 – 8.4	0.30/0.58
Lindsay 1	26.6	39.1	13.2	10.4 – 16.1	0.31
Lindsay 2	26.6	35.7	12.1	9.5 – 14.7	0.31
Lindsay 3	26.6	44.4	32.5	28.0 – 36.2	0.31
Verulam 1	26.4	25.7	18.9	16.3 – 21.0	0.33
Verulam 2	26.4	33.1	24.2	20.9 – 27.0	0.31
Verulam 3	26.4	36.3	26.6	22.9 – 29.7	0.31
Verulam 4	26.4	40.3	29.5	25.5 – 32.9	0.31
Bobcaygeon	26.3	44.6	32.7	28.1 – 36.4	0.31
Gull River	26.5	52.8	38.7	33.3 – 43.1	0.32
Shadow Lake	25.7	38.0	27.8	24.0 – 31.0	0.30
Gneiss	27.3	52.6	16.2	11.8 – 21.5	0.28

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.5-3: Stability Requirements for RB and Containment Common Mat Foundation**

Load Combination	Overturning	Sliding	Flotation
$D + H + W$	1.5	1.5	
$D + H + E'$	1.1	1.1	
$D + F'$			1.1
<i>where</i>			
$D$ = Dead Load, $W$ = Wind			
$H$ = Lateral soil pressure, $E'$ = Design Basis Earthquake			
$F'$ = Buoyant forces of design basis flood			

**Note:**

If quasi-static method using the maximum force effects from the SSI analysis results is used for seismic stability evaluations, the minimum factor of safety against sliding and overturning is no less than 1.25 in accordance with Clause 5.9 of CSA N289.3.

**Table 3.5-4: Acceptance Criteria for SCCV**

(a) Allowable Stress/Strain Limits for Factored Loads

Material	Force Classification	Type of Force Action	Criteria for Factored Loads	
			Stress Limit	Strain Limit, if any
Concrete	Primary	Membrane	$0.60f_c'$	-
		Membrane + Bending	$0.75f_c'$	-
	Primary + Secondary	Membrane	$0.75f_c'$	-
		Membrane + Bending	$0.85f_c'$	0.002
Steel Plates	Primary	Membrane or Membrane + Bending	$0.90F_y$	-
	Primary + Secondary	Membrane or Membrane + Bending	-	$2\varepsilon_y^*$

\* Limit for mechanical (net) strain, calculated by subtracting strain induced by secondary force from total strain.

(b) Allowable Stresses for Service Loads

Material	Force Classification	Type of Force Action	Criteria for Service Loads
			Stress Limit
Concrete	Primary	Membrane	$0.30f_c'$
		Membrane + Bending	$0.45f_c'$
	Primary + Secondary	Membrane	$0.45f_c'$
		Membrane + Bending	$0.60f_c'$
Steel Plates	Primary	Membrane or Membrane + Bending	$0.50F_y$
	Primary + Secondary	Membrane or Membrane + Bending	$0.67F_y$

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.5-5: Acceptance Criteria for Containment Closure Head**

Service Level	Acceptance Criteria <sup>*1</sup>			
	$P_m$	$P_L$	$P_L + P_b$ <sup>*2</sup>	$P_L + P_b + Q$
Test Condition	$0.8 S_y$	$1.15 S_y$	$1.15 S_y$	N/A <sup>*3</sup>
Design Condition	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	N/A <sup>*3</sup>
Level A	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	$3.0 S_m$
Level C	$1.2 S_{mc}$ or <sup>*4</sup> $1.0 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	N/A <sup>*3</sup>
Level D	$S_f$	$1.5 S_f$	$1.5 S_f$	N/A <sup>*3</sup>

\*1: Acceptance Criteria is defined by ASME BPVC, Subsection NE Subarticles NE-3221.1 through 3221.4.

$P_m$  = primary stress: general membrane.

$P_L$  = primary stress: local membrane.

$P_b$  = primary stress: bending.

$Q$  = secondary stress: membrane plus bending.

$S_y$  = material's yield strength at temperature as in ASME BPVC Section II, Part D (Reference 3.5-44), Table Y-1.

$S_m$  = allowable stress intensity  $S_m$  is the value given in ASME BPVC Section II Part D, Subpart 1, Tables 2A and 2B.

$S_{mc}$  = allowable stress intensity  $S_{mc}$  is 1.1 times the  $S$  listed in ASME BPVC Section II Part D, Subpart 1, Tables 1A and 1B, except  $S_{mc}$  shall not exceed 90% of the material's yield strength at temperature shown in ASME BPVC Section II, Part D, Subpart 1, Tables Y-1.

$S_f$  = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII, ASME BPVC Code Section III (Reference 3.5-45). In the application of Appendix XXVII,  $S_m$ , if applicable, is as specified in NE-3112.4(a)(1).

\*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Subarticle NE-3221.3(d) for other than a solid rectangular section.

\*3: N/A = Not applicable. No evaluation required.

\*4: The larger of the two values listed is chosen as a limit load.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.5-6: Acceptance Criteria for Other MC Components**

Service Level	Acceptance Criteria <sup>*1</sup>			
	$P_m$	$P_L$	$P_L + P_b$ <sup>*2</sup>	$P_L + P_b + Q$
Test Condition	$0.8 S_y$	$1.15 S_y$	$1.15 S_y$	N/A <sup>*3</sup>
Design Condition	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	N/A <sup>*3</sup>
Level A, B	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	$3.0 S_m$
Level C	$1.2 S_{mc}$ or <sup>*4</sup> $1.0 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	N/A <sup>*3</sup>
Level D	$S_f$	$1.5 S_f$	$1.5 S_f$	N/A <sup>*3</sup>
Post-flooding Condition	$1.2 S_{mc}$ or <sup>*4</sup> $1.0 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	$1.8 S_{mc}$ or <sup>*4</sup> $1.5 S_y$	$3.0 S_m$

\*1: Acceptance Criteria for other than Post-flooding Condition is defined by ASME BPVC, Subsection NE Subarticles NE-3221.1 through 3221.4. For Post-flooding Condition, Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress, per item 5 of SRP Acceptance Criteria in U.S. NUREG-0800 SRP 3.8.2.

\*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Subarticle NE-3221.3(d) for other than a solid rectangular section.

\*3: N/A = Not applicable. No evaluation required.

\*4: The larger of the two values listed is chosen as a limit load

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

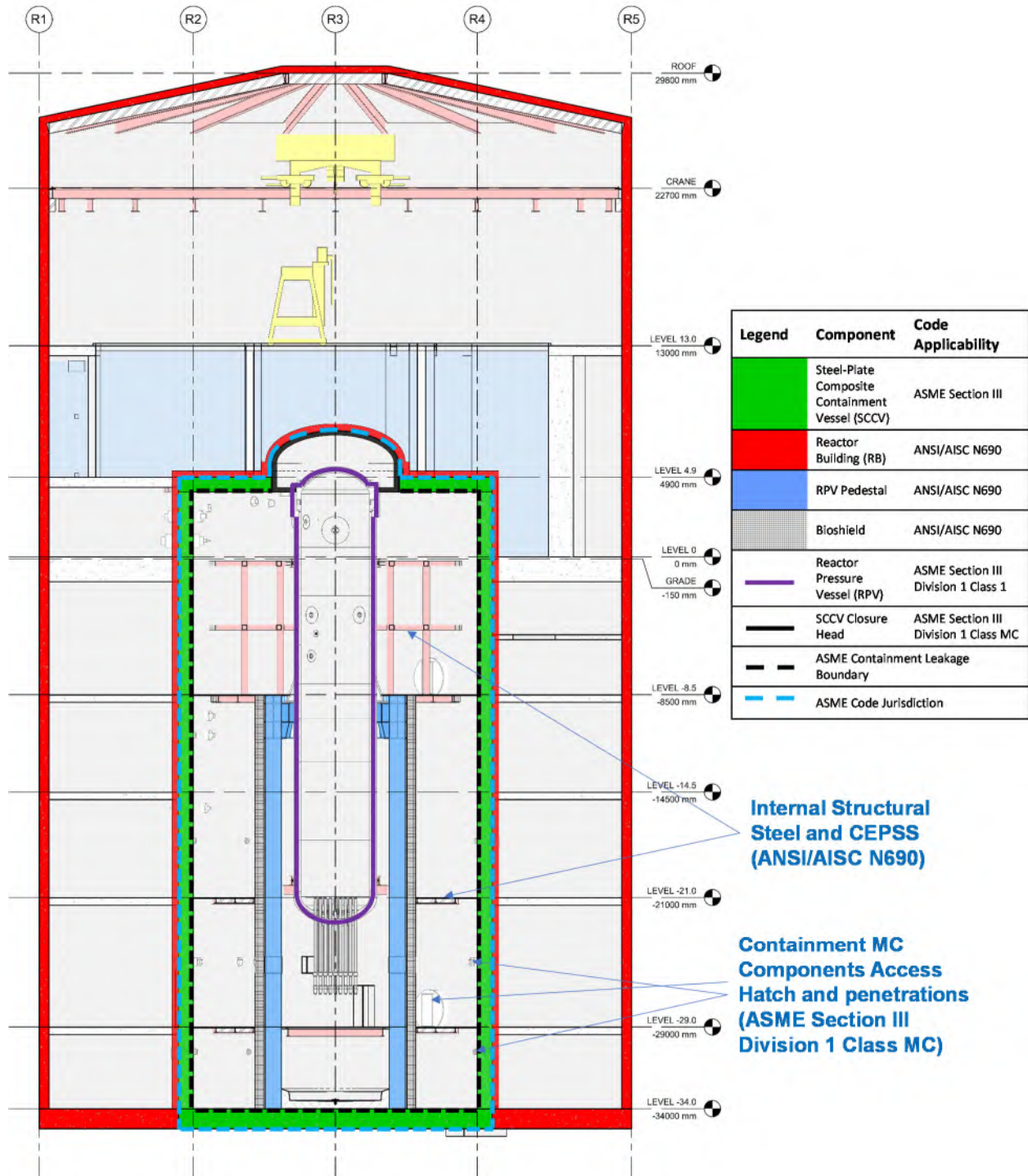
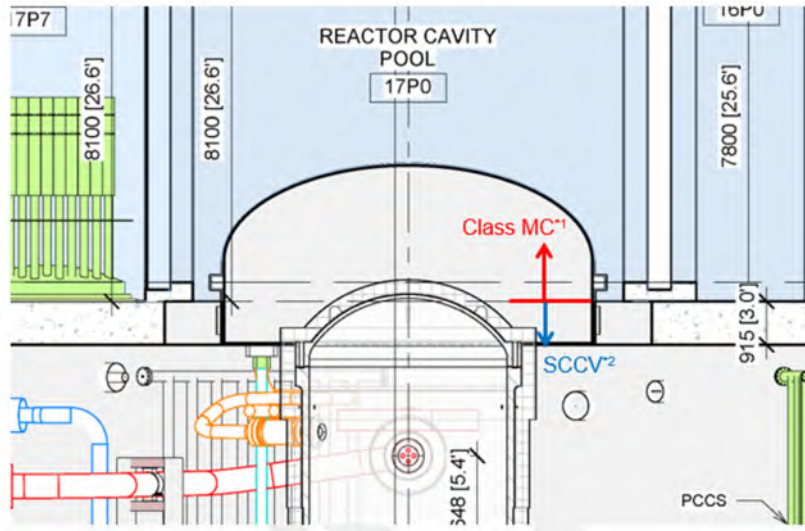


Figure 3.5-1: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 3.5-2: Containment Closure Head Structure Boundary**

\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

\*2: Is designed in accordance with NEDC-33926P

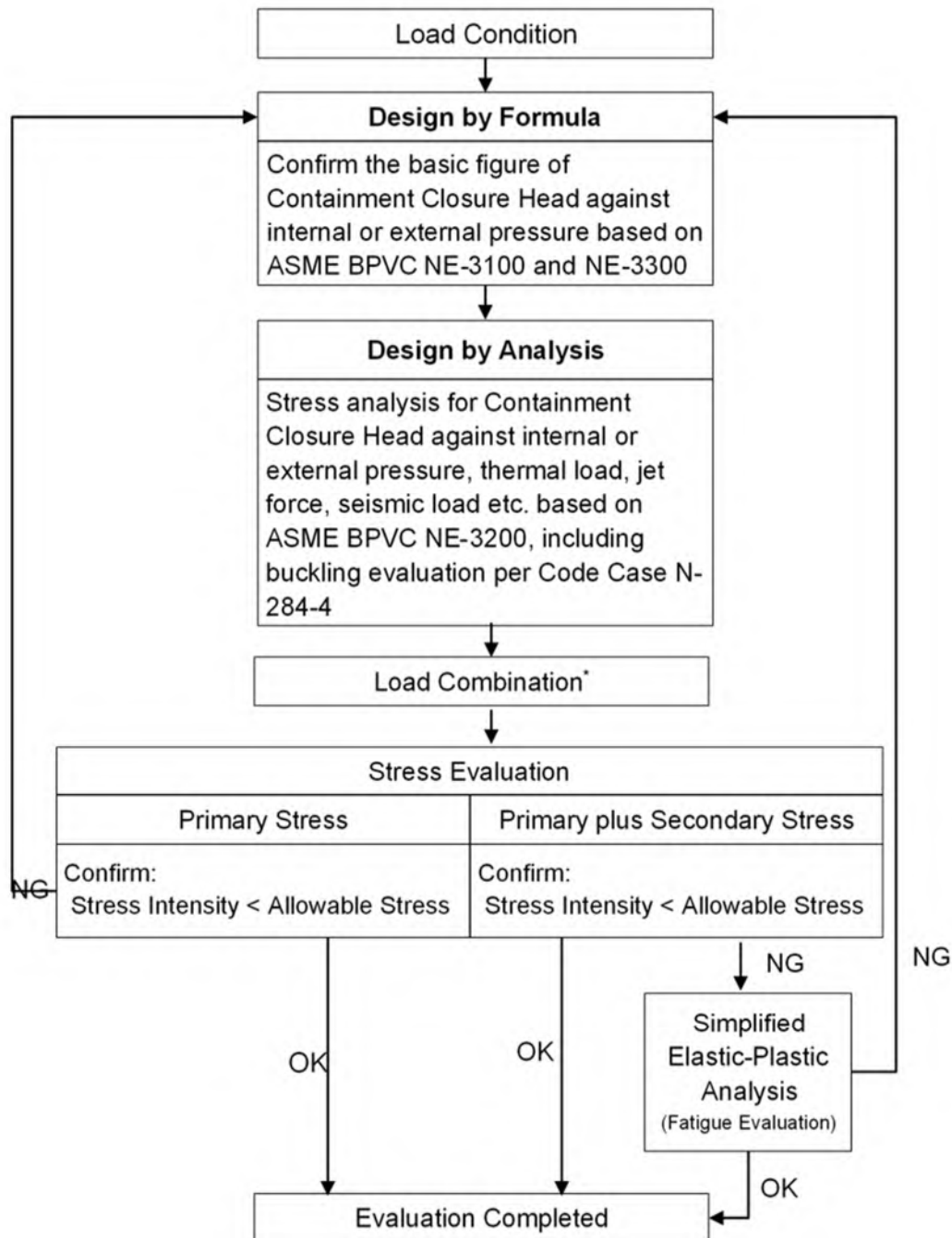
\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

3-176



\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

\*2: Is designed in accordance with NEDC-33926P



**Figure 3.5-5: Design Procedures for the Containment Closure Head**

\*: Steel Portion: U.S. NRC RG 1.57 and U.S. NUREG-0800 SRP 3.8.2

Concrete Portion: NEDC-33926P

\*: Steel Portion: US NRC RG 1.57 and U.S. NUREG-0800 SRP 3.8.2  
Concrete Portion: NEDC-33926P

### **3.6 General Design Aspects for Mechanical Systems and Components**

Section 3.6 provides the general design aspects used for safety class and non-safety class mechanical systems and components. It includes special considerations for mechanical components, dynamic testing and analysis of structures, systems, and components, required codes for ASME BPVC Section III Division 1 Class 1, 2, and 3 components, and component supports, including core support structures. In addition, general design aspects for Control Rod Drive System, Reactor Vessel Internals, system piping, and threaded fasteners are presented. Further, this section discusses the functional design, qualification and in-service testing program requirements for pumps, valves, and dynamic restraints.

Chapter 1 provides the codes and standards and editions that are applicable to the design of mechanical systems and components and is used as input to Section 3.6.

Sections 3.1 and 3.2 are used as input to Section 3.6 and provide the general design principles, criteria, and classification used for design of mechanical systems and components. Among these principles are design for robustness, reliability, and fail-safe operation. Additionally, the systems and components are required to be redundant, diverse, independent, separate and of supply quality commensurate with the safety classification, seismic category, and supply category. The design and qualification of mechanical components is performed using a graded approach with the highest level of rigor applied to Safety Class 1 (SC1) components.

Subsection 3.3.1 develops the seismic input criteria and building spectra used as input to Section 3.6 for seismic qualification of Seismic Category B active mechanical components and system functionality. Additionally, Seismic Category A passive mechanical component supports, and equipment supports use the seismic spectra for qualification.

Section 3.9 provides the equipment qualification requirements including environmental, dynamic, functional qualification, and Electromagnetic Compatibility (EMC), which are used as input to Section 3.6.

#### **Codes and Standards Used in the Design of Mechanical Systems and Components**

ASME BPVC Section III Division 1, ASME B31.1 (Reference 3.6-10), and ASME B31.3 (Reference 3.6-12) are applied for the design of mechanical systems, components and piping including piping components.

Table 3.6-1 provides the pressure boundary codes and standards utilized in the BWRX-300 mechanical system and component design.

#### **Mechanical Equipment Separation for Safety Class 1**

Mechanical equipment separation measures for the BWRX-300 contribute to system reliability in the performance of any Safety Category 1 function including (but not necessarily limited to) interconnecting piping, valves, and associated mechanical controls and instrumentation. Additionally, where necessary adjacent systems are considered in mechanical equipment separation (as related to human factors, mechanical maintenance, and seismic interaction).

Principles of physical separation include:

- A. Separation by geometry (layout, distance, orientation, elevation, and including separate structures)
- B. Separation by barriers (e.g., walls, shields), both vertical and horizontal
- C. Separation by a combination of (A) and (B)

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Per CNSC REGDOC-2.5.2, Section 7.4.1 (Reference 3.6-16), the plant design takes into account the potential for internal hazards such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures are provided to ensure that nuclear safety is not compromised.

Per CNSC REGDOC-2.5.2, Section 7.6.1.1, vertical separation, or other protection is provided where physical separation by horizontal distance alone may not be sufficient for some common cause failures such as flooding.

Defense Line (DL) functions that mitigate the same event are independent from each other to the extent practicable. All PIEs with a frequency greater than 1E-05 can be mitigated by functions in DL3 and separately by functions in either DL2 or 4a. Therefore, SSC performing DL3 functions are separate, to the extent practicable, from SSC that perform Safety Category functions in DL2 and DL4a. Separation is also provided between redundant SSC that perform DL3 functions (Safety Category 1) to the extent practicable.

The redundancy methods are used to protect from Single Active Failures or events; examples include utilization of safety class structures, spatial separation, three-hour rated fire barriers, and isolation devices.

The application of the single failure criterion to fluid systems is described in Subsection 3.1.7.5.

Separation of components may be by physical distance or by barriers. An example is the provision of principal fire barriers to delineate individual fire zones; such barriers may also serve as barriers to other hazards, as per CNSC REGDOC-2.5.2 Section 7.6.1.1.

The following SC mechanical equipment items are considered:

- Piping Systems
- Valves
- Rotating Equipment
- Vessels
- Ductwork Systems
- Instrumentation

### **Piping Systems**

Piping systems include piping to and from SC and SCN SSC. These include their connected bellows, mechanical connections, support guides, and structural supports. They may include wall or floor sleeves and penetrations, pipe fittings including wells and branch connections, structural restraints (and appurtenances), and attached sampling. Piping systems also include vent/drain/test/flush/clean-out taps including closures, instrument sensing line piping or tubing and instrument racks. Finally, they also include pneumatic or hydraulic system tubing, manifolds and controls appurtenances.

### **Valves**

Valves include those that control fluid flow to and from SC and SCN SSC. Valves include the valve body assembly, actuators, appurtenances, and all non-electrical connections.

### **Rotating Equipment**

Rotating equipment includes pumps, fans and compressors, gear sets or power coupling subsystems, and electric motors or other rotary-power driven subsystems. Their components include rotating casing, including base, frame, supports and drive.

### **Vessels**

Vessels include heat exchangers and tanks, including their supports, filter assemblies, and nozzles.

### **Ductwork Systems**

Ductwork systems include:

- Duct runs
- Active and pre-set dampers
- Fire dampers
- Screens
- Vents/reliefs/blow out panels
- Filters or air filtration assemblies/subsystems

### **Instrumentation**

Instrumentation includes:

- Mechanically activated instruments used to monitor reactor and plant processes
- The associated non-electrical transmission
- Sensors
- Actuator systems
- In-line instruments with associated taps

### **Zone of Influence**

The degree and type of separation required varies with the following potential hazards in a power plant zone:

1. **Missiles** - A missile is an unrestrained mass with sufficient kinetic energy to cause damage to the safety systems or required safety components. Definition of missile and missile protection requirements are addressed in Subsection 3.3.5
2. **Pipe Whip** - Pipe whip is usually consequent to a pipe failure resulting in a complete segment separation break. The area in the vicinity of the postulated break of high-energy piping is defined as the pipe whip damage zone. Pipe whip protection requirements are addressed in Subsection 3.4.4.
3. **Fluid Jet** - The fluid jet is usually consequent to a high-energy pipe break but may also be the result of intentional equipment action. Jet impingement protection requirements are addressed in Subsection 3.4.4

### **Fire Area and Fire Zone**

A fire area is an area sufficiently bounded to withstand the hazards associated with the fire area and, as necessary, to protect important equipment within the fire area from a fire outside the area. A fire zone, however, is a subdivision of fire area(s) for analysis purposes that is not necessarily bound by fire-rated barriers.

Fire zone protection requirements are addressed in Chapter 9A, Section 9A.6. Separation of vulnerable mechanical equipment from areas containing significant combustible materials is provided by fire barrier materials or housings, fire-rated walls or doors (including consideration for ductwork isolations), barrier piping around processes containing flammable or combustible fluids to isolate the hazard, and in certain locations by atmospheric inerting (oxygen concentration suppression below combustible level or replacement with nitrogen, such as in containment).

### **Flood Zone**

Internally generated flooding may occur by pipe or tank failure, fire suppression system operation, misaligned systems with openings in the affected zone, maintenance errors, or failure of a drainage system. Flood protection requirements are addressed in Subsections 3.3.3.1 and 3.4.2.

Separation by flood hazard containment walls, dikes, curbs, trenches or pits, watertight doors, elevated equipment mounting location (mezzanine or different floor) or pedestals or placing vulnerable equipment in watertight housings may be used.

### **Design Load and Load Combination for Mechanical Systems and Components**

Design loads and loading combinations are based on normal operation and off-normal operation. Subsection 3.6.1.1 below provides the operational transients, resulting loads, and load combinations.

Design loads and load combinations for fixed mechanical equipment are provided in Table 3.6-2. Fixed equipment includes the mechanical, electrical, and instrument components, and the component housings and structural supports that are anchored to civil structure(s) but are not a part of the civil structure itself, such as mechanical or electrical penetrations. Examples include the reactor pressure vessel (RPV), RPV Internals, RPV supports, instrumentation, piping, electrical equipment, and the component supports.

A discussion of plant normal and off-normal operation can be found in Chapter 1, Section 1.8, and Chapter 6, Sections 6.2 and 6.4.

### **Design for System Duty of Mechanical Systems Based on Event Frequencies**

Table 3.6-3 is used as a general event list for all hardware system duty design specifications. Events are mainly classified into:

- Design Condition 1 (DC-1): Normal Planned Operation
- Design Condition 2 (DC-2): Anticipated Operational Occurrence
- Design Condition 3 (DC-3): Design Basis Accident
- Design Condition 4 (DC-4): Design Extension Condition

The BWRX-300 utilizes the four Service Levels used in the ASME Code, Levels A, B, C and D, as well as testing conditions, in the design of fixed equipment. The design basis specifies the capabilities that are necessary for the plant in various operational states.

Conservative design measures and sound engineering practices are applied in the design basis for plant states. This approach provides a high degree of assurance that no significant damage

will occur to the reactor core, and that radiation doses will remain within established regulatory limits.

### **3.6.1 Special Topics for Mechanical Components**

This subsection addresses information concerning methods of analysis for components and supports.

#### **3.6.1.1 Computer Programs Used in Analyses**

The major computer programs used in the mechanical system and component analyses of the major safety class components are described in Chapter 3, Appendix 3C .

The computer programs used in the analyses of Seismic Category A and B components are maintained either by General Electric Company (GE) or by outside computer program developers.

The GEH Software is controlled under NEDO-11209-A (Ref. 3.6-17). CSA N286.7 (Ref. 3.6-14) is used to determine acceptability of code use for the BWRX-300 in Canada. In either case, the quality of the programs and the computed results are controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature.

#### **3.6.1.2 Operational Transients, Resulting Loads and Load Combinations**

The plant duty cycles represent transient conditions that are used for development of the BWRX-300 system and component design during Normal Operation, Anticipated Operational Occurrence (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs), which are Beyond Design Basis Events. Requirements are evaluated for the system design and performance as it relates to complete reactor operation. The duty is recorded as inputs to the system design for each specific primary and auxiliary hardware system. Duty can be defined from a pressure and temperature perspective, mostly when variations in either variable are expected in important locations for the reactor.

The number of cycles associated with each event for the design of the Reactor Pressure Vessel (RPV), Reactor Coolant Pressure Boundary (RCPB), and other ASME pressure boundary components designed for fatigue are listed in Table 3.6-9. Tables 3.6-4 through Table 3.6-8 break down the operational cycles by plant condition. The plant operating conditions are identified as normal, AOO, DBA, DEC, or testing as defined in Subsection 3.6.3.2. Appropriate Service Levels (A, B, C, D, or testing), as defined in the ASME BPVC, are designated for design limits. The design and analyses of ASME Class piping and equipment using specific applicable thermal-hydraulic transients, which are derived from the system behavior during the events listed in Table 3.6-3, are documented in the design specifications and/or stress reports of the respective equipment. Table 3.6-2 shows the load combinations and the standard acceptance criteria for ASME Section III components. Tables 3.6-10, 3.6-11, and 3.6-12 provide the specific load combinations and acceptance criteria for piping systems.

#### **3.6.1.3 Experimental Stress Analysis**

Experimental stress analysis methods are used in compliance with the provisions of ASME BPVC Section III Division 1, Mandatory Appendix II (Reference 3.6-9). ASME Class 1 and some ASME Class 2 mechanical components that require both functionality and adequate structural capacity during seismic events, are laboratory tested in accordance with CSA N289.4 (Reference 3.6-13) and ASME Standard QME-1 (Reference 3.6-20) as discussed in Subsection 3.9.3.2.1.



#### **3.6.1.4 Considerations for the Evaluation of Fault Conditions**

All equipment designed to ASME BPVC Section III Division 1 is evaluated for the faulted (Service Level D) loading conditions. In all cases, the calculated actual stresses are compared to the allowable ASME BPVC Section III Division 1 Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

##### ***3.6.1.4.1 Fine Motion Control Rod Drive***

The Fine Motion Control Rod Drive (FMCRD) major components that are part of the RCPB are analyzed and evaluated for the ASME Service Level D faulted conditions in accordance with the ASME BPVC Section III Division 1, Subsection NB (Reference 3.6-3). Refer to Chapter 4, Subsection 4.6.2.1.1 for FMCRD mechanism details.

##### ***3.6.1.4.2 CRD Hydraulic Control Unit***

The Hydraulic Control Unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests that are part of the seismic and dynamic qualification program establish the loads in the horizontal and vertical directions as the HCU capability for the frequency range that is likely to be experienced in the plant. These tests also ensure that the reactor trip function of the HCU can be performed under these loads. Dynamic analysis of the HCU with the mounting beams is performed to assure that the maximum faulted condition loads remain below the HCU capability. Refer to Chapter 4, Subsection 4.6.2.1.3 for HCU details.

##### ***3.6.1.4.3 Reactor Pressure Vessel Assembly***

The design of the RPV assembly, out to and including the integral Reactor Isolation Valves (appurtenances), RPV Top Head, and housings for FMCRD and in-core Nuclear Instrumentation complies with Subsections NB and NG of the ASME BPVC Section III Division 1 as applicable. For faulted conditions, the reactor vessel is evaluated using elastic analysis.

Elastic analysis methods and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary, Seismic Category B, ASME BPVC Section III, Division 1, Class 1 valves. The ASME BPVC Section III Division 1 allowable stress is applied to assure integrity under applicable loading conditions including faulted condition. The functional qualification of the Reactor Isolation Valves (RIVs), is analyzed and/or tested for seismic and other dynamic conditions.

##### ***3.6.1.4.4 Core Support Structures and Other Safety Class Reactor Internal Components***

The core support structures, the internal portion of Nuclear Instrument and CRI housings, and other safety class reactor internal components are evaluated for faulted conditions. The basis for determining the faulted loads for seismic events and other dynamic events is given in Subsection 3.6.2.3 and Subsection 3.6.2.2, respectively. The allowable Service Level D limits for evaluation of these structures are per ASME BPVC Section III Division 1, Service Level D equations.

For the shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

##### ***3.6.1.4.5 RPV Stabilizers, Reactor Skirt and FMCRD Housing and Nuclear Instrumentation Housing Restraints (Supports)***

The calculated maximum stresses to meet the allowable stress limits are based on the ASME BPVC Section III Division 1, Subsection NF (Reference 3.6-7), for the RPV stabilizer, RPV skirt

and supports for the FMCRD housing and Nuclear Instrumentation housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building Vibration events.

**3.6.1.4.6 Reactor Isolation Valves, and Other ASME BPVC Section III Division 1 Class 1 and 2 Valves**

Elastic analysis methods and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary, Seismic Category B, ASME BPVC Section III Division 1 Class 1 and 2 valves. The ASME BPVC Section III Division 1 allowable stresses are applied to assure integrity under applicable loading conditions including faulted condition. The functional qualification of the major active valves, including Reactor Isolation Valve (RIVs), Containment Isolation Valves (CIVs), ICS Purge valves, and ICS Condensate Return valves are analyzed and/or tested for seismic and/or other dynamic conditions.

**3.6.1.4.7 Fuel Storage and Refueling Equipment**

The fuel storage and fuel handling equipment is described in detail in Section 9A.1. This includes the Fuel Pool structure, Fuel Racks, Fuel Cooling system, and Fuel Handling Equipment.

CNSC REGDOC 2.5.2 Section 6.2, Subsection 7.3.4.1, and Subsection 8.12.2, require that the same Section 3.1 fundamental safety functions as those that apply to the Reactor be utilized for fuel storage and handling. Due to physical and structural separation, Safety Class equipment cannot be affected by a fuel handling accident.

A summary of the design considerations used to establish nuclear criticality safety under all operational and faulted (ASME Service Level D) conditions is described below.

All fuel storage racks are designed and qualified to operate within their performance requirements under the anticipated ranges of the normal, abnormal or accident plant environments and are designed to withstand a Design Basis Earthquake (DBE) without failure of the basic structure or damage to the active region of irradiated fuel.

**3.6.1.4.8 Fuel Assembly (Including Channel)**

The Fuel Assembly including channel is described in detail in Section 4.2.3.

The channel is subjected to mechanical tests to demonstrate the adequacy of the GNF2 channel for seismic/dynamic loads. The channel was tested to determine the allowable bending load that could be sustained without buckling or collapsing the channel.

The Fuel Assemblies are designed for worst-case conditions that evaluate maximum stresses, fatigue, control rod insertion, fretting, corrosion/hydriding, and compatibility/dimensional changes. The results of the testing and analysis requires that the safety class components maintain the required functionality and structural capacity during ASME Level D service conditions.

**3.6.1.4.9 ASME BPVC Section III Division 1 Class 2 and 3 Vessels**

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from Articles NCD-3300 and NCD-3200 of the ASME BPVC Section III Division 1 Subsection NCD (Reference 3.6-4).

**3.6.1.4.10 ASME BPVC Section III Division 1 Class 2 and 3 Pumps**

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from Article NCD-3400 the ASME BPVC Section III Division 1 Subsection NCD.

#### **3.6.1.4.11 ASME BPVC Section III Division 1 Class 2 and 3 Valves**

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2 and 3 valves. The equivalent allowable stresses for valves using elastic techniques are obtained from Article NCD-3500 of the ASME BPVC Section III Division 1 Subsection NCD.

#### **3.6.1.4.12 ASME BPVC Section III Division 1 Class 1, 2 and 3 Piping**

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from Article NB-3600 (for Class 1 piping) of the ASME BPVC Section III Division 1 Subsection NB and Article NCD-3600 (for Class 2 and 3 piping) of the ASME BPVC Section III Division 1 Subsection NCD.

#### **3.6.1.4.13 Inelastic Analysis Methods**

Inelastic analysis is only applied to BWRX-300 components to demonstrate the acceptability of two types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. These two events are as follows:

- Postulated gross piping failure
- Postulated blow out of a Control Rod Drive housing caused by a weld failure

The design criteria for pipe failure effects and mitigating features are provided in Subsection 3.4.4.1. Except for the analysis of pipe failures, inelastic methods are not used in BWRX-300 piping design.

The mitigation of the CRDH attachment weld failure relies on components with regular functions to mitigate the weld failure effect. The components are specifically:

- Core support plate
- Control Rod Guide Tube
- CRD Housing
- Control Rod Drive (CRD) outer tube
- Bayonet Fingers

Only the bodies of the CRGT, CRDH, and CRD outer tube are analyzed for energy absorption by inelastic deformation.

### **3.6.2 Dynamic Testing and Analysis of Systems, Components, and Equipment**

This Subsection 3.6.2 presents the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to fluid flow and postulated seismic events. Structural requirements for conduits and cable tray supports and Heating, Ventilation and Air Conditioning duct supports are developed as discussed in Subsection 3.6.2.5.7.

#### **3.6.2.1 Piping Vibration, Thermal Expansion and Dynamic Effects**

The overall test program is divided into two phases:

1. Pre-operational test phase
2. Initial startup test phase

Piping vibration, thermal expansion, and dynamic effects testing is performed during both of these phases. Discussed below are the general requirements for this testing. It is noted that because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow.

#### ***3.6.2.1.1 Vibration and Dynamic Effects Testing***

The purpose of these tests is to confirm that the piping, components, restraints, and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady-state Flow Induced Vibration (FIV) and anticipated operational transient conditions.

#### ***3.6.2.1.2 Seismic Qualification of Safety Class Mechanical Equipment***

Section 3.9 provides methodology for qualification of SC1 Mechanical equipment.

#### ***3.6.2.1.3 Tests and Analysis Criteria and Methods***

Section 3.9 provides tests and analysis criteria methods.

### **3.6.2.2 Qualification of Safety Category Mechanical Equipment**

The following subsections discuss the testing or analytical qualification of the safety class major mechanical equipment, and other ASME BPVC Section III Division 1 equipment including equipment supports.

#### ***3.6.2.2.1 CRD and CRDH***

The qualification of the CRDH (with enclosed FMCRD) is done analytically, and the stress results of the analysis establish the structural integrity of these components. Dynamic tests are conducted to verify the operability of the CRD during a dynamic event. A simulated test, imposing dynamic deflection in the fuel channels up to values greater than the expected seismic response, is performed.

The correlation of the test with analysis is via the channel deflection, not the housing structural analysis, because insert ability is controlled by channel deflection, not housing deflection.

#### ***3.6.2.2.2 Core Support (Fuel Support and Control Rod Guide Tube)***

A detailed analysis imposing dynamic effects due to seismic and other RBV events is performed to show that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

#### ***3.6.2.2.3 CRD Hydraulic Control Unit***

The HCU is analyzed for the seismic and other RBV loads in the faulted condition and the maximum stress on the HCU frame is calculated to be below the maximum allowable for the faulted condition.

#### ***3.6.2.2.4 Fuel Assembly (Including Channel)***

The Fuel Assembly (including channel) qualification for seismic and faulted load conditions is described in Chapter 4, Subsections 4.2.2 and 4.2.3.

#### ***3.6.2.2.5 Containment Isolation Valves and Reactor Isolation Valves***

The CIVs for main steam and other process system piping that penetrates containment, and RIVs are qualified for seismic and other RBV loads. The fundamental requirement following a Design Basis Earthquake (DBE) or other faulted RBV loadings is to close and remain closed after the event. This capability is demonstrated by the test and analysis.

#### **3.6.2.2.6 Other ASME BPVC Section III Division 1 SSCs**

Other equipment, including associated supports, is qualified for seismic and other RBV loads to ensure its functional integrity during and after the dynamic event. The equipment is tested, if necessary, to ensure its ability to perform its specified function before, during, and following a seismic event.

Dynamic load qualification is done by testing, analysis, or both as described in Section 3.9.

Refer to Section 3.9 for additional information on the dynamic qualification of valves.

#### **3.6.2.2.7 Supports**

Analyses or tests are performed for component supports to assure their structural capability to withstand seismic, faulted, and other dynamic excitations. Pre-qualified manufactured standard component supports, or engineered component supports that are qualified to specified required service levels for seismic, faulted, and dynamic excitation do not require additional analyses or testing.

#### **3.6.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions**

The major reactor internal components within the vessel are subjected to extensive testing, coupled with dynamic system analyses, to properly evaluate the resulting FIV phenomena during normal reactor operation and from anticipated operational transients.

##### **3.6.2.3.1 Initial Startup Flow Induced Vibration Testing of Reactor Internals**

A reactor internals vibration measurement and inspection program is conducted only during initial startup testing. These reactor internal inspections and tests consist of evaluating Flow Induced Vibrations, including any flow excited acoustic and structural resonance that is detected in initial startup testing. Analytical thermal-hydraulic fluid models are developed that replicate plant startup conditions to predict resonance effects on the reactor internals. These predictive models are used in design to eliminate undesired acoustics and structural resonances to a practical extent.

##### **3.6.2.3.2 Initial Startup Testing**

Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Steady-state and transient conditions of natural circulation flow operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals.

##### **3.6.2.3.3 Dynamic System Analysis of Reactor Internals Under Faulted Conditions**

The loads to the Reactor Internals that occur because of faulted events and the deterministic analyses performed to determine the response of the reactor internals are as follows:

- Reactor Internal Pressures
- External Pressure and Forces on the Reactor Vessel
- LOCA Loads
- Seismic Loads

##### **3.6.2.3.4 Correlations of Reactor Internals Vibration Tests with the Analytical Results**

Prior to initiation of the instrumented vibration measurement program for a prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these

analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been used in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

### **3.6.3 Codes for ASME BPVC Section III Division 1, Class 1, 2 and 3 Components, Component Supports and Core Support Structure**

Subsection 3.6.3 discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of the ASME BPVC Section III Division 1.

The ASME BPVC Section III Division 1, Section III, requires that a design specification be prepared for ASME BPVC Section III Division 1 Class 1, 2 and 3 components. The design specifications for ASME BPVC Section III Division 1 Class 1, 2 and 3 components, supports, and appurtenances are prepared under administrative procedures that meet the ASME BPVC Section III Division 1 rules. The specifications conform to and are certified to the requirements of the applicable subsection of the ASME BPVC Section III Division 1. The ASME BPVC Section III Division 1 also requires design reports for Class 1, 2 or 3 components be prepared which demonstrate that the as-built components satisfy the requirements of the respective ASME design specification for each component and the applicable ASME BPVC Section III Division 1. These design specifications and the design reports are completed by the licence applicant, or the applicant's authorized agent, in accordance with the responsibilities outlined under the ASME BPVC Section III Division 1. The ASME BPVC Section III Division 1 design reports include the record of as-built reconciliations, for example, the evaluations of changes to piping support locations, the pre-operational testing, and results, and reported construction deviation resolution, and includes the small-bore piping analysis.

#### **3.6.3.1 Loading Combinations, Design Transients and Stress Limits**

Subsection 3.6.3.2 delineates the criteria for selection and definition of design limits and loading combinations associated with Normal Operation, Anticipated Operational Occurrence (AOO), Design Basis Accidents (DBAs), Design Extended Conditions (DECs) and specified seismic and other RBV events for the design of safety ASME BPVC Section III Division 1 components (except containment components which are discussed in Section 3.5).

This section discusses the ASME BPVC Section III Division 1 Class 1, 2, and 3 equipment and associated pressure-retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME BPVC Section III Division 1 Class 1, 2 and 3 equipment are covered in Subsections 3.6.1, 3.6.3.6 and 3.6.3.7. Seismic-related loads and dynamic analyses are discussed in Subsection 3.3.1. Table 3.6-9 presents the plant events to be considered for the design and analysis of all BWRX-300 ASME BPVC Section III Division 1 Class 1, 2, and 3 components, component supports, equipment, and core support structures per ASME BPVC Section III Division 1 Subsection NG (Reference 3.6-8). Specific loading combinations considered for evaluation of specific equipment are derived from Table 3.6-2 and are contained in the design specifications and design reports for the respective equipment. For Class 1 components where analysis for

cyclic operation is evaluated in accordance with ASME BPVC Section III Division 1 subarticle NB-3222.4, the fatigue usage evaluation includes the use of environmental fatigue curves.

Specific load combinations and acceptance criteria for Class 1 piping are shown in Table 3.6-10. Also, for Class 1 piping, the operating temperatures above ambient or below ambient are included in the fatigue analysis. The installation temperature state for the piping system is defined as a temperature of 21 C for Class 1, 2, 3 or ASME B31.1 piping.

The design life for the BWRX-300 Standard Plant is 60 years. A 60-year design life is a requirement for all major plant components. Additional life is added for components required during decommissioning. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable. The design life requirement allows for refurbishment and repair, as appropriate, to assure that the design life of the overall plant is achieved.

### **3.6.3.2 Events Considered in Evaluating Effect of Loads on Fixed Equipment**

All events that the BWRX-300 might credibly experience during a reactor-year are evaluated in Chapter 15, to establish the plant design basis, including plant fixed equipment. The associated loads and duty cycles associated with each event are considered in combination with additional events in load combinations as applicable. These event combinations are divided into the four plant conditions with associated frequency of occurrence and ASME BPVC Section III Division 1 design levels.

The following are the plant condition events and transients associated with the BWRX-300 design:

#### ***3.6.3.2.1 Normal Operation***

Normal planned operation is operation under any condition permitted within specified Operational Limits and Conditions (OLCs) irrespective of the anticipated frequency of occurrence of that condition, which is planned and deliberate and not in specific response to Postulated Initiating Events (PIEs). Normal planned operations include startup, power operation, shutting down, shutdown, maintenance, testing, and refueling.

Adequate evaluation of normal operation loads includes loads due to dead weight, temperature, prestress, pressure, fluid flow (including FIV when applicable), thermal and fluid reaction forces and other loads due to moving parts within a component or system. Such loads are considered in the design, installation, and mounting, of equipment and components.

#### ***3.6.3.2.2 Anticipated Operational Occurrences***

Anticipated Operational Occurrences (AOO) are those operating transient events that are expected to occur more frequently than 1E-02 per reactor-year. Chapter 15, Subsection 15.5.3 provides event analyses of Level B PIE AOOs.

Adequate evaluation of associated loads, load combinations, and duty cycles of the AOO transient effects are considered in the design, installation, and mounting, of equipment and components.

#### ***3.6.3.2.3 Design Basis Accident Events***

Design Basis Accidents (DBA) are those events with frequencies of occurrence between 1E-02 to 1E-05 per reactor-year DBAs are mitigated by Defense Line 3. Chapter 15, Subsection 15.5.4 provides event analyses of Level C PIE DBAs.

#### ***3.6.3.2.4 Design Extension Condition Events (DEC)***

Design Extension Conditions (DEC) are events that are less frequent than 1E-05 reactor-year. DEC event analyses demonstrate the capability of the plant to cope with scenarios involving

Defense Line 3 Common Cause Failures (CCFs) and provide a systematic evaluation of potential cliff-edge effects outside the plant design bases. DEC transient events are mitigated by SSC associated with Defense Line 4a and DL2 functions that are unaffected by the PIE and additional failures identified in the event sequence. Chapter 15, Subsections 15.5.5 through 15.5.9 provides event analyses of Level D PIE DEC.

#### **3.6.3.2.5 Seismic Events**

Seismic design parameters and associated seismic events defined in Subsection 3.3.1 are used in qualification of mechanical system components. The magnitude of seismic events is determined by Ground Response Spectra accelerations applied to Building Structures and creating Amplified Response Spectra (ARS) accelerations at various building elevations where the components are located. These ARS accelerations are used in qualification of Mechanical systems and equipment and a determination of component and system structural and/or functional capacity is determined. Seismic Category A (passive components) require only structural code adequacy. Seismic Category B (active components) such as valves and pumps require both structural code adequacy and functional capacity under seismic demand. Chapter 3, Subsection 3.9.3 provides seismic qualification methodology to assure both component structural and/or functional capacity under seismic operational conditions are met.

The seismic categorization of SSC is defined in Section 3.2 and related to the seismic category to the more general safety strategy defense lines. In summary, Defense Lines 3 and 4b are generally Seismic A or B and Defense Line 4b also has an additional requirement of satisfying the plant-level High Confidence of Low Probability of Failure (HCLPF) criteria.

#### **3.6.3.2.6 Non-LOCA Fault**

Non-LOCA Fault consists of any DEC event not considering a LOCA which has a significantly low frequency of occurrence to be considered as a faulted event.

#### **3.6.3.2.7 Plant Testing**

Plant testing events are occasional operating loads imposed during pre-operational testing or periodic operational testing.

### **3.6.3.3 Classification of Components**

All SSC of the BWRX-300 design are designated by Safety Class, Quality Group, and Seismic Category according to guidance in Section 3.2 which are consistent with their Defence-in-Depth categorization defined in the BWRX-300 Safety Strategy, in Section 3.1. Appendix 3A provides the Classification Table for Plant SSC.

### **3.6.3.4 Establishment of Design, Service, and Test Loadings and Limits**

Design, Service, and Test Loadings and Limits for fixed equipment components and supports are in accordance with ASME BPVC Section III Division 1 (Reference 3.6-5).

For IEEE Equipment, SC1 electrical equipment is evaluated with respect to the load combinations in this document using IEC/IEEE 60980--323 and IEC/IEEE 60980--344 Acceptance Criteria, Codes and Standard (References 3.6-18 and 3.6-19).

For SC1, actuators and power operated valve assemblies are evaluated with respect to the load combinations in this document in accordance with the provisions of ASME Standard QME-1.



### **3.6.3.5 Acceptance Criteria**

Components and supports comply with the design rules established for design, service, and test loadings in the appropriate with the appropriate subsection of the ASME BPVC, Section III, Division 1 (References 3.6-1 through 3.6-8).

Design documentation is completed in accordance with the requirements of the Subsection of the ASME BPVC applicable to the component or support.

### **3.6.3.6 Loading Criteria**

#### **3.6.3.6.1 Loading Conditions**

The loadings that are considered in designing a component include, but are not limited to, those in (a) through (g) below:

- a. Internal and external pressure
- b. Impact loads, including rapidly fluctuating pressures
- c. Weight of the component and normal contents under operating or test conditions
- d. Superimposed loads such as other components, operating equipment, insulation, corrosion resistant or erosion resistant linings, and piping
- e. Wind loads, snow loads, vibrations, and earthquake loads, where specified
- f. Reactions of supporting lugs, rings, saddles, or other types of supports
- g. Temperature effects

As appropriate ASME BPVC, Division 1, Section III, Paragraph, NB-3111, NCD-3111, NE-3111, NF-3111 or NG-3111, is applied for a complete list of required load conditions to consider.

Consistent with the ASME BPVC Section III Division 1, the stresses resulting from differential anchor movements during dynamic events are considered secondary stresses.

#### **3.6.3.6.2 Design Loadings**

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3112, NCD-3112, NE-3112, NF-3112 or NG-3112, as applicable.

#### **3.6.3.6.3 Service Conditions**

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3113, NCD-3113, NE-3113, NF-3113 or NG-3113, as applicable.

Each service condition to which the components may be subjected is classified in accordance with Service Limits designated in the Component Design Specifications in such detail as will provide a complete basis for design, construction, and inspection.

For ASME BPVC Section III Division 1, Class 1 Components, the requirements of (1) and (2) below apply.

1. Level B Conditions. The estimated duration of service conditions for which Level B Limits are specified are included in the Design Specifications.
2. Level C Conditions. The total number of postulated occurrences for all specified service conditions for which Level C Limits are specified are limited to no more than 25 stress cycles having a  $S_a$  value greater than that for  $10^6$  cycles from the applicable fatigue design curves of Section III Appendices, Mandatory Appendix I.

When the Component Design Specification requires computations to demonstrate compliance with specified Service Limits, the Component Design Specification provides information from which Service Loadings can be identified (pressure, temperature, mechanical loads, cycles, or transients).

**Design Pressure** - The specified internal and external Design Pressure is not to be less than the maximum difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the most severe loadings for which the Level A Service Limits are applicable.

The Design Pressure includes allowances for pressure surges.

**Design Temperature** - Except as otherwise defined in ASME BPVC, Division 1, NB-3112 for Class 1 components, the specified Design Temperature is not less than the expected maximum mean metal temperature through the thickness of the part considered for which Level A Limits are specified.

**Design Mechanical Loads** - The specified Design Mechanical Loads are in accordance with NCA-2142.1C.

#### **3.6.3.6.4 Test Loadings**

**Test Pressure** - The specified internal and external test pressures are as required by the ASME BPVC, Section III, Division 1.

**Test Loads** - Loads due to other types of required tests are included as required by the ASME BPVC, Section III, Division 1.

**Test Temperature** - Test temperature is defined to ensure that thermal effects are considered in test loads.

#### **3.6.3.7 Loading Phenomena**

Section 3.6.3.7 describes the types of load phenomena, that is considered for components, as applicable.

##### **3.6.3.7.1 Flow Induced Vibration**

Flow of fluids past objects creates local pressure disturbances, which exert forces on the object. These forces can cause dynamic responses depending on the forcing function and dynamic characteristics of the object. Flow induced vibrations have been noted in nuclear power plant systems, which produce vortex shedding (e.g., heat exchangers), pump (reciprocating or centrifugal), and thermodynamic instability conditions. Design changes are reviewed for potential FIV mechanisms, evaluating all modes of system operation including both normal and abnormal conditions. Requirements for vibration monitoring are not within the scope of this document.

FIV loads may be associated with Service Level A for those structures (e.g., reactor internals) where the loads exist during normal operation. For FIV loads associated with transients that are not considered part of normal operation, the FIV loads are evaluated as part of the alternative service level.

#### **Vortex Shedding**

Vortex shedding occurs at certain fluid velocities when a fluid flows past objects. The dynamic response is controlled by proper spacing of the support plates for the tube bundle. The vibration cannot be eliminated but it can usually be controlled. It is important that these cases consider all potential modes of component operation. Vortex shedding hydrodynamic mass effects are

considered. Other components susceptible to flow induced vibration are pressure, flow, and temperature sensors, which encroach upon the flow stream.

### **Pressure Fluctuations**

Pressure fluctuations in a vapor or gas-state fluid (e.g., steam) occur due to flow past branch piping connections and branch connected components (e.g., safety valve “bell chamber” resonance), flow through short radius elbow fittings that induce flow separation effects, flow passing through valve chambers, flow past sharp-edged in-line pipe components (e.g., orifices, weld joint backing rings, valve seat rings), or two or more individual flows entering a common header or drum that generates an acoustic response. These various flow disturbances generate acoustic waves that can travel forward and backward in a piping system. If of sufficient strength and at a component’s susceptible frequency, these acoustic resonances can cause cyclic fatigue and result in component failure.

Pumps create pressure fluctuations in a fluid system. In most system designs, these fluctuations are insignificant. However, the possibility exists that these fluctuations, coupled with unintentional but improper system or component structural characteristics, can cause resonant vibrational response in the system or component. Component structural characteristics are designed to assure a resonance value sufficiently high to avoid excitation by evaluated system fluid fluctuations. Pressure attenuation devices are used as applicable to significantly reduce the effects of this phenomenon.

### **Thermodynamic Instability**

Under certain system design features and operating modes, fluid dynamic forces can be generated, which create large pressure variations. These have been noted in certain feedwater systems where a relatively cold fluid layer is in contact with a relatively hot steam region; under certain operating modes significant water-hammer-type phenomena have occurred causing a breach of the pressure-retaining boundary.

#### **3.6.3.7.2 Rapid Valve Closure or Opening**

Extremely rapid valve closure or opening in a fluid system can create large pressure waves which can propagate through a piping system and into connected components. This rapid motion could be caused by operating characteristics of the valve (e.g., stiffness of diaphragm in pneumatic operators), the fluid flow forces acting on the valve parts during all modes of operations.

For example, TSV closure has been identified as being capable of generating large pressure waves which could cause significant dynamic response. Prior to TSV closure, saturated steam flows through main steam piping at nuclear boiler rated pressure and mass rate. Steam flow to the turbine comes to a stop at the instant the turbine stop valve closes. The flow of steam travels in the main steam line through the vessel nozzle and into the vessel. This results in a compressive acoustic load on steam dryer outer hood, as well as steam impingement load on steam dryer outer hood. Additionally, repeated reflections of the compression wave in the main steam line generate time-varying forces in the main steam piping. System, components, and structures in the Reactor Building, Steam Tunnel and Turbine Building may be affected.

#### **3.6.3.7.3 Isolation Condenser Operation**

The thermal effects associated with operation of ICS and the loads such as pressure resulting from operation of ICS are considered. Loads associated with the breaks of ICS high pressure lines in the pool are considered. The major loads imposed on ICS result from:

- Sudden reactor isolation at power operating conditions
- During station blackout (i.e., unavailability of all alternate current power)

- Failure to Scram
- LOCA

#### **3.6.3.7.4 Failures of High-Energy Fluid System Piping**

The effects of postulated pipe breaks in high-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

#### **3.6.3.7.5 Failures of Moderate-Energy Fluid System Piping**

The effects of postulated pipe cracks in moderate-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

#### **3.6.3.7.6 Fuel Lift Loads**

Fuel lift is the postulated process under which a combination of vertical motion of the RPV support, scram uplift forces on the fuel assemblies and vertical hydraulic forces result in fuel assemblies lifting off from their seating surfaces on the fuel support. The reaction load of the fuel on the core support structures is considered.

#### **3.6.3.8 Safety Class Functional Criteria**

For any normal or off-normal design condition event, safety class equipment and piping can accomplish the safety class functions as required by the event and incurring no permanent changes that could deteriorate the ability to accomplish safety class functions as required by any subsequent design-condition event.

For any emergency or faulted design-condition event, safety class equipment, and piping are capable of accomplishing their safety class functions as required by the event, but repairs could be required to ensure their ability to accomplish safety class functions as required by any subsequent design-condition event.

#### **3.6.3.9 Reactor Pressure Vessel Assembly**

The reactor vessel assembly includes: the RPV pressure boundary out to and including the nozzles, the RIV's, and the housings for FMCRD and nuclear instrumentations. The RPV assembly is an ASME BPVC Section III, Division 1, Class 1.

The feedwater nozzle design does not allow incoming feedwater flow to have direct contact with the nozzle bore region. A double thermal sleeve design provides protection against thermal cycling on the nozzle bore. The ICS Condensate Return nozzles use a similar single thermal sleeve design to mitigate thermal cycling of the nozzle bore during initial IC train operation when accumulated condensate is draining.

The stress analysis is performed on the RPV for various plant operating conditions (including faulted conditions) by using elastic methods, except as noted in Subsection 3.6.1.4.3. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are provided in Table 3.6-2.

The RPV internals are classified in Chapter 3, Section 3.2, and Appendix 3A. Complete stress reports on these components are prepared in accordance with the ASME BPVC Section III, Division 1, requirements.

#### **3.6.3.10 Main Steam Piping**

The MS piping trains extending from the outboard MSRIV to and including Seismic Interface Restraints (SIR) that are outboard of the MSCIVs are designed and constructed in accordance with the ASME BPVC Section III Division 1 rules for Class 2 Nuclear Components. Stresses are

calculated on an elastic basis for each service level and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. Table 3.6-11 shows the specific load combinations and acceptance criteria for Class 2 piping that apply to this piping.

The MSCIVs, are designed and constructed in accordance with the ASME BPVC III Division 1, NCD-3500 requirements for Class 2 components.

The MS system piping extending from the outboard SIR to the turbine stop valve is constructed in accordance with the ASME B31.1 Criteria.

### **3.6.3.11 Other Components**

#### **3.6.3.11.1 Isolation Condenser System (ICS) Condenser and Piping**

The ICS piping inside the primary containment between the RPV and the Isolation Condenser Heat Exchanger is designed and constructed in accordance with the ASME BPVC Section III Division 1 requirements for Class 1 piping. The isolation condenser and piping outside containment are designed and constructed in accordance with ASME BPVC Section III Division 1 Class 2 requirements.

#### **3.6.3.11.2 CUW System Heat Exchangers**

The CUW heat exchangers (regenerative) are not part of a safety system. However, the heat exchangers are Seismic Category NS equipment. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the CUW System heat exchanger components.

#### **3.6.3.11.3 SDC System Pump and Heat Exchangers**

The SDC heat exchangers (nonregenerative) are not part of a safety system. However, the pumps and heat exchangers are Seismic Category NS equipment respectively. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the SDC System pump and heat exchanger components.

#### **3.6.3.11.4 ASME BPVC Section III Division 1, Class 2 and 3 Vessels**

ASME BPVC Section III Division 1, Class 2 and 3 vessels are constructed in accordance with the ASME BPVC Section III Division 1. The analysis of these vessels is performed using elastic methods.

#### **3.6.3.11.5 ASME BPVC Section III Division 1, Class 1, 2 and 3 Valves**

ASME BPVC Section III Division 1, Class 1, 2, and 3 valves are constructed in accordance with the ASME BPVC Section III Division 1.

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The analysis of these valves is performed using elastic methods. Refer to Subsection 3.6.3.9 for additional information on valve operability.

#### **3.6.3.11.6 ASME BPVC III Division 1, Class 1, 2 and 3 Piping**

ASME BPVC Section III Division 1, Class 1, 2 and 3 piping is constructed in accordance with the ASME BPVC Section III Division 1. For ASME BPVC Section III Division 1, Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME BPVC Section III Division 1, and fatigue usage is determined. For ASME BPVC Section III Division 1, Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. If a NB-3600 analysis is performed for ASME BPVC Section III Division 1, Class 2 or 3 pipe, all analyses required for ASME BPVC Section III Division 1, Class 1 pipe as specified in this document and the ASME

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

BPVC is performed. Tables 3.6.10 and 3.6.11 shows the specific load combinations and acceptance criteria for ASME BPVC Section III Division 1, Class 1, 2, and 3 piping systems.

### **3.6.3.12 Valve Operability Assurance**

This subsection discusses operability assurance of active ASME BPVC Section III Division 1 valves, including actuators (Refer to Subsection 3.9.6.2).

Valves that perform an active Safety Category 1 function are functionally qualified to perform their required functions. For valve designs developed for the BWRX-300 that were not previously qualified, the qualification programs meet the requirements of ASME QME-1 (For valve designs previously qualified to standards other than ASME QME-1), the following approach is used:

1. Qualification specifications (e.g., design specifications) consistent with Appendices QV-I and QV-A of QME-1 are prepared to ensure the operating conditions and safety class functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility.
2. Suppliers are required to submit, for review and approval, application reports, as described in QME-1, that describe the basis for the application of specific predictive methods and/or qualification test data to a valve application.
3. The application reports provided by the suppliers are reviewed for adherence to specification requirements to ensure the methods used are applicable and justified and to verify any extrapolation techniques used are justified. A gap analysis is performed to identify any deviations from QME-1 in the valve qualification. Each deviation is evaluated for impact on the overall valve qualification. If the conclusion of the gap analysis is that the valve qualification is inadequate, then the valve may be qualified using a test-based methodology, as allowed by QME-1.

Functional qualification addresses key lessons learned from industry efforts, particularly on air- and motor-operated valves, many of which are discussed in Section QVG of QME-1. For example:

1. Evaluation of valve performance is based on a combination of testing and analysis, using design similarity to apply test results to specific valve designs.
2. Testing to verify proper valve setup and acceptable operating margin is performed using diagnostic equipment to measure stem thrust and torque, as appropriate.
3. Sliding friction coefficients used to evaluate valve performance (e.g., disk-to-seat friction coefficients for gate valves and bearing coefficients for butterfly valves) account for the effects of temperature, cycle history, load, and internal parts geometry.
4. Actuator sizing allows margin for aging/degradation, test equipment accuracy and other uncertainties, as appropriate.
5. Material combinations that may be susceptible to galling or other damage mechanisms under certain conditions are not used.

Subsection 3.9 provide details on the seismic qualification of valves and on the Environmental Qualification of valves.

The major safety class active valves are the RIVs, Condensate Return Valves and CIVs. These valves are designed to meet the ASME BPVC Section III Division 1 BPVC requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is provided individually below.

### **3.6.3.13 Main Steam Containment Isolation Valves**

The MSCIVs are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a Design Basis Accident (DBA) and DBE.

### **3.6.3.14 Other Active Valves**

Other safety class active valves are ASME BPVC Section III Division 1 Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves operate during a dynamic seismic and other RBV event.

#### **3.6.3.14.1 Procedures**

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components, which are depended upon to cause the valve to accomplish its intended function, are developed to assure these functions are accomplished.

#### **3.6.3.14.2 Tests**

Prior to installation of the SC1 valves, the following tests are performed at the factory facility as required in the field:

- Shell hydrostatic test to the ASME BPVC Section III Division 1 requirements
- Seat leakage tests
- Obturator hydrostatic test
- Functional tests to verify that the valve opens and closes within the specified time limits when subject to the design differential pressure

The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

#### **3.6.3.14.3 Check Valves**

Due to the simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

- Stress analysis including the dynamic loads where applicable
- In-shop hydrostatic tests
- In-shop seat leakage test

### **3.6.3.15 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation**

A practical problem arises in attempting to describe tests for simple devices (e.g., relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a simple device, that is an integral part of an assembly, may be subjected to the same dynamic load tests while in an operating condition. Thus, the performance of a simple device may be monitored during the test. However, for complex panels, such a test is not always practical. In this situation, the following alternate approach may be followed.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar but inoperative devices installed, is vibration tested to determine if the panel response accelerations. Installing the non-operating devices assures that the test panel has representative structural characteristics of a production

panel. The accelerations are measured by accelerometers installed at the device attachment locations. The accelerations are less than the levels at which the devices were qualified. If the acceleration levels at all the device locations are found to be less than the levels to which the devices are qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices are requalified to the higher levels.

#### **3.6.3.16 Design of Pressure Relief Devices**

The NBS system does not utilize safety or relief valves for overpressure relief. During normal operation, the mainsteam flow to the turbine is throttled to control system pressure. Chapter 6, Section 6.2 describes the method of overpressure relief.

#### **3.6.3.17 Component Supports**

The establishment of the design/service loadings and limits is in accordance with the ASME Section III, Division 1, Article NCA-2000 and Subsection NF. These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events. The combination of loadings and stress limits are included in the Design Specification of each component and support.

ASME Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops, Pipe whip restraints are not considered as pipe supports.

The design of bolts for component supports is specified in the ASME BPVC III Division 1, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

#### **3.6.3.18 Piping Supports**

Supports and their attachments for ASME BPVC Section III Division 1 Class 1, 2, and 3 piping are designed in accordance with Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed as specified in Section 3.5.

The design of supports for the non-nuclear piping satisfies the requirements of ASME B31.1 Power Piping Code, Paragraphs 120 and.

#### **3.6.3.19 Reactor Pressure Vessel Stabilizer**

The RPV stabilizer is designed as a SC1 linear type component support in accordance with the requirements of ASME BPVC Section III Division 1 Subsection NF. The stabilizer provides a reaction point near the upper end and lower end of the RPV to resist horizontal loads caused by effects such as earthquake, pipe rupture, and RBV. The design loading conditions, and stress criteria and the calculated stresses will meet the ASME BPVC Section III Division 1 allowable stresses in the critical support areas for various plant operating conditions.

#### **3.6.3.20 Floor-Mounted Major Equipment**

The condenser modules in the Isolation Condenser System (ICS) are analyzed to verify the adequacy of their support structure under various plant operating conditions. The analysis applies



the maximum shear, moment, and accelerations calculated from the seismic response analysis for the Reactor Building at the attachment locations on the pool floor for the ICS.

In the ICS module analysis, no credit is taken for damping effects of the pool water. Additionally, the mass of the condensers is increased by an amount equivalent to the weight of water they displace. This conservative factor accounts for the hydrodynamic effects that include impulsive loads and convective loads (sloshing of the pool water).

In all cases, the load stresses in the critical support areas of the ICS modules are maintained within ASME BPVC Section III Division 1 allowable.

#### **3.6.3.21 Other ASME BPVC Component Supports**

The ASME BPVC Section III Division 1 component supports and their attachments (other than those discussed in the preceding subsection) are designed in accordance with ASME BPVC Section III Division 1, Subsection NF up to the interface with the building structure. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Table 3.6-2. Active component supports are discussed in Subsection 3.6.3.18. The stress limits are per ASME BPVC Section III Division 1, Subsection NF, and NB-3600 and NCD-3600. The supports are evaluated for buckling in accordance with ASME BPVC Section III Division 1.

### **3.6.4 Control Rod Drive System**

The CRD system consists of mechanical components that provide the means for movement of the control rods. The CRD system provides one of the independent reactivity control systems. The control rods and the drive mechanisms are capable of reliably controlling reactivity changes either under conditions of AOOs, or under DBA conditions. A positive means for inserting the rods is always maintained to ensure appropriate margin for malfunction, such as stuck rods. Because the CRD system is a safety class system and portions of the CRD system are a part of the RCPB, the system is designed, fabricated, and tested to quality standards commensurate with the safety class functions to be performed. This provides an extremely high probability of accomplishing the safety class functions either in the event of AOOs or in withstanding the effects of DBAs and natural phenomena such as earthquakes.

The CRD system includes the FMCRD mechanisms, the HCU assemblies, and the CRD hydraulic system. The system extends inside the RPV to the coupling interface with the control rod blades.

#### **3.6.4.1 Descriptive Information on Control Rod Drive System**

Descriptive information on the FMCRDs as well as the entire CRD system is contained in Chapter 4, Subsection 4.6.

#### **3.6.4.2 Applicable Control Rod Drive System Design Specification**

The CRD system, which is designed to meet the functional design criteria outlined in Chapter 4, Subsection 4.6.1, consists of the following:

- Electro-hydraulic fine motion control rod drive
- Hydraulic Control Unit (HCU)
- Hydraulic pumps
- Electric power supply R20 system to the FMCRD motors – CRD Boundary is at the motor
- Interconnecting piping

- Flow control valves
- Instrumentation

Those components of the CRD system forming part of the primary pressure boundary are designed according to ASME BPVC Section III Division 1 BPVC, Class 1 requirements.

The quality group classification of the components of the CRD system is outlined in Appendix 3A and are designed to the codes and standards in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRD system components are discussed in the following locations: transients in Chapter 3, Subsections 3.6.1.1, 3.6.3.6 and 3.6.3.7, faulted conditions in Chapter 3, Subsections 3.6.1.4.1 and 3.6.1.4.2, and seismic testing in Chapter 4, Subsections 4.6.1 and 4.6.2.

### **3.6.4.3 Design Loads and Stress Limits**

#### **3.6.4.3.1 Allowable Deformations**

The ASME BPVC Section III Division 1, Subsection NB components of the CRD system are evaluated analytically and the design loading conditions, and stress criteria are as given in Table 3.6-2.

### **3.6.5 Reactor Pressure Vessel Internals**

Reactor pressure vessel internals are described in Chapter 5, Section 5.4.

### **3.6.6 Functional Design, Qualification and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints**

Chapter 3, Section 3.9, Equipment Qualification provides the methodology for qualification of Pumps and Valves. The qualification involves both determining component functionality while maintaining structural integrity. Seismic testing of components is performed as well as use of analytical methods.

Chapter 3, Subsection 3.6.3.17 discusses methodology for qualification of dynamic restraints.

In-service Testing Programs are developed for required operability and functional tests for components as described in Chapter 3, Subsection 3.10.3.

### **3.6.7 Piping Design**

The design of safety class piping systems, piping components and pipe supports is based on the code rules established under the ASME BPVC Section III, Division 1 code for Class 1, Class 2, and Class 3 nuclear piping, components and supports. For non-ASME Code class components, ASME B31.1 power piping, and ASME B31.3 process piping codes are used. Safety classifications of safety, seismic categories, and quality groups for piping SSCs are established within the system chapters. The simplified schematic diagrams within the system chapters identify the system safety class, seismic class, and quality boundaries. The functional, operational, and safety requirements are unique to each system and the required loading conditions are applied as specified in the specific ASME Code class sections.

#### **3.6.7.1 ASME Class 1 Piping Design Rules and Analysis**

ASME Class 1 piping design conforms to the requirements of ASME BPVC Section III Division 1 Paragraph NB code rules that covers both piping and piping components. The pipe supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC Section III Division 1, Paragraph NF. The anchor sleeve of the containment structure penetrations meets the requirements of ASME BPVC Section III Division 1, Paragraph NE (Reference 3.6-6).

#### **3.6.7.1.1 Overpressure Protection**

The details and certification of Overpressure Protection design for each piping system are in the System Overpressure Protection Reports.

#### **3.6.7.1.2 Boundaries**

The boundaries of the Class 1 piping in each system are outlined in the system Piping and Instrumentation Diagrams (P&IDs).

Support design jurisdictional boundaries at interfaces between piping and structure by intervening elements that are defined per ASME BPVC Section III Division 1 - Subsection NF – Supports, Subarticle NF-1130. If piping supports transmit loads to surface-mounted baseplates as discussed in Subparagraph NF-1132(d), the baseplates are within the building structure jurisdiction.

Where ASME BPVC Section III Division 1 Class 2 piping is connected to ASME BPVC Section III Division 1 Class 1 piping, the rules for expansion and flexibility for A ASME BPVC Section III Division 1 Class 1 piping applies out to the first anchor in the ASME BPVC Section III Division 1 Class 2 piping system. However, the resulting solution of forces and moments are used to evaluate stresses in accordance with the allowable criterion of ASME BPVC Section III Division 1 Subarticle NCD-3650.

#### **3.6.7.1.3 Classifications**

##### **Code Classification**

Piping that is classified as Quality Group A meets the requirements for ASME BPVC III Division 1 Class 1 components provided in ASME BPVC Section III Sub Article NB-3600.

The pipe supports attached to Quality Group A piping meet the appropriate requirements of ASME BPVC Section III Paragraph NF.

##### **Seismic Classification**

Seismic categories are to be in accordance with those listed on the system P&ID.

##### **Energy (High/Moderate) Classification**

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Section 3.4.4.2 for further explanation.

#### **3.6.7.1.4 Material Requirements**

The material properties used in Class 1 analyses is in accordance with ASME BPVC Section II – Materials – Part D – Properties (Metric).

##### **Examination and Repair**

The examination and repair of all Class 1 materials and welds is performed using the methods and acceptance standards as specified in ASME BPVC Section III Subarticle NB-2500.

In-service inspection requirements for Class 2 and 3 piping and components are defined in Subsection 3.10.5.

##### **Fracture Toughness Requirements**

Pressure-retaining ferritic material, and material welded thereto are impacted tested in accordance with the requirements of NB-2300 and NB-2400 to ensure adequate fracture toughness properties.

#### **3.6.7.1.5 Design Conditions**

##### **Design Service Life**

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years. Additional time in-service for startup and decommissioning activities is included as applicable.

##### **Design Pressure and Temperatures**

The design pressures and temperatures of each piping system are identified in the respective system design documentation.

##### **Design Duty Cycles**

The pressure-temperature duty cycles to be used in the fatigue analysis are specified in the respective system Pressure-Temperature Duty Cycle drawings. Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

##### **Environmental Conditions**

All SC1 piping, and components, are capable of performing their safety class functions when exposed to specified environmental conditions specified in the Environmental Qualification Envelope. Piping system active components are environmentally qualified as specified in Subsections 3.9.3 and 3.9.4.

#### **3.6.7.1.6 Test Loads**

The only test loads on the piping system are due to hydrostatic testing. The loads due to hydrostatic testing are in accordance with NB-6000.

#### **3.6.7.1.7 Static Loads**

##### **Pressure**

The design pressure and operating pressure for each system/component are as specified in the respective system design documentation.

##### **Weight**

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe are considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and/or refueling outage are designed to accommodate the increased weight.

##### **Thermal Expansion**

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases shall be established to account for various operating conditions and for calculating the range of thermal expansion stresses between all pairs of load sets.

The installation temperature for the piping systems is defined as a temperature of 21° C for Class 1 piping unless basis is provided to use a higher temperature. The ambient state shall be included as an analysis load set with defined cycles.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design.

### **Thermal Attenuation/Stratification**

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

#### **3.6.7.1.8 Dynamic Loads**

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement). Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Loss-of-Coolant Accident Loads
- Reactor Pressure Vessel and Containment Isolation Valve Transients
- Thermal Stratification

#### **3.6.7.1.9 Plant Events and Load Combinations**

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

Load combinations and acceptance criteria for the BWRX-300 Class 1 piping are provided in Table 3.6-10.

#### **3.6.7.1.10 Analytical Computer Codes Used for Piping Stress, Component Stress, and Support Structural Qualifications**

Chapter 3, Appendix 3C provides a listing of and description of applicable safety computer codes used for qualification of piping, mechanical components, and pipe supports.

#### **3.6.7.1.11 Analysis Methodology and Stress Reports**

Piping system stresses shall be calculated on an elastic basis for each service level.

For ASME BPVC Section III Division 1 Class 1 piping systems and components, stress reports are prepared in accordance with ASME BPVC Section III Division 1 Class 1 requirements and include applicable equipment qualification reports for active components.

### **3.6.7.2 ASME BPVC Section III Division 1 Class 2/3 Piping Design Rules and Analysis**

ASME BPVC Section III Division 1 Class 2/3 piping design conforms to the requirements of ASME BPVC Section III Division 1 Subsection NCD that covers both piping and piping components. Load combinations and acceptance criteria for the BWRX-300 Class 2 piping are provided in Table 3.6-11.

The containment penetration sleeve of ASME Class 2 piping is an anchor for the piping. The sleeve of the containment structure penetrations meets the requirements of ASME BPVC Section III Division 1, Subsection NE (Reference 3.6-6).

#### **3.6.7.2.1 Overpressure Protection**

The details and certification of Overpressure Protection design for each piping system are in the System Overpressure Protection Reports.

#### **3.6.7.2.2 Boundaries**

The boundaries of the Class 2 and 3 piping in each system are outlined in the system P&IDs and simplified diagrams shown in the system PSAR chapters.

Support design jurisdictional boundaries at interfaces with piping, structure, or intervening elements are defined in ASME BPVC Section III Division 1, Subsection NF-1130. If piping supports transmit loads to surface-mounted baseplates as discussed in Subsection NF-1132(d), the baseplates are within the building structure jurisdiction.

#### **3.6.7.2.3 Classifications**

##### **Code Classifications**

Detailed classifications of pipe and components are defined in the system design documents. Piping that is classified as ASME BPVC Section III Division 1 Class 2 or ASME BPVC Section III Division 1 Class 3 meet the requirements for ASME BPVC Section III Division 1 Class 2 and 3 components provided in Subsection NCD-3600 of the ASME Code.

Where ASME BPVC Section III Division 1 Class 2 piping is connected to ASME BPVC Section III Division 1 Class 1 piping, the rules for expansion and flexibility for Class 1 piping apply out to the first anchor in the ASME BPVC Section III Division 1 Class 2 piping system. However, the resulting solution of forces and moments are used to evaluate stresses in accordance with the allowable criterion of NCD-3650.

The pipe supports attached to the ASME BPVC Section III Division 1 Class 2 and 3 piping meet the appropriate requirements of Subsection NF of the ASME Code.

##### **Seismic Classification**

Seismic categories are to be in accordance with those listed on the system design documents.

##### **Energy (High/Moderate) Classification**

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Subsection 3.4.4.2 for further explanation.

#### **3.6.7.2.4 Materials**

##### **Material Specifications**

The material properties used in Class 2 or 3 analyses are in accordance with ASME BPVC Section II – Materials – Part D – Properties (Metric) (Reference 3.6-1).

##### **Examination and Repair**

The examination and repair of all Class 2 and 3 materials and welds are performed using the methods and acceptance standards as specified in NCD-2500.

In-service inspection requirements for Class 2 and 3 piping and components are defined in Subsection 3.10.5.

##### **Fracture Toughness Requirements**

Pressure-retaining ferritic material, and material welded thereto are impact tested in accordance with the requirements of NCD-2300 and NCD-2400 to ensure adequate fracture toughness properties.

#### **3.6.7.2.5    *Design Conditions***

##### **Design Service Life**

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years plus any additional time in-service for startup and decommissioning activities as applicable.

##### **Design Pressures and Temperatures**

The design pressures and temperatures of each piping system are identified in the respective system design documents.

##### **Design Duty Cycles**

Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

##### **Environmental Conditions**

All SC1 piping, and components, are capable of performing their Safety Category functions when exposed to the environmental conditions.

#### **3.6.7.2.6    *Design Input Loads***

##### **Test Loads**

The only test loads on the piping system are due to hydrostatic testing. The loads due to hydrostatic testing are in accordance with NCD-6000.

##### **Static Loads**

##### **Pressure**

The design pressure and operating pressure for each system/component are as specified in the respective System Line list.

##### **Weight**

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe are considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and/or refueling outage are designed to accommodate the increased weight.

##### **Thermal Expansion**

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses.

The installation temperature for the piping systems is defined as a temperature of 21 °C for Class 2 and 3 piping.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design. Thermal anchor movements of less than 1.6 mm are considered negligible and do not need to be considered in the analysis.

### **Thermal Attenuation/Stratification**

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

#### **3.6.7.2.7 Dynamic Loads**

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement).

Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Loss-of-Coolant Accident Loads
- Turbine Stop Valve Closure
- Reactor Pressure Vessel and Containment Isolation Valve Transients

#### **3.6.7.2.8 Plant Events and Load Combinations**

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

### **Load Combinations**

The load combinations and acceptance criteria in Table 3.6-11 are applicable to all ASME BPVC Section III Division 1 Class 2 and 3 piping systems, structures, and components.

#### **Load Combinations for Piping and Components**

The load combinations and acceptance criteria in Table 3.6-11 are applied to the analysis of ASME BPVC Section III Division 1 Class 2 and 3 piping systems and components.

#### **3.6.7.3 ASME B31.1 Piping Design Rules and Analysis**

Non-Safety class power piping conforms to ASME B31.1 code.

Load combinations and acceptance criteria for the BWRX-300 Class 1 piping are provided in Table 3.6-12.

Each Non-Safety class power piping systems includes the piping, pipe supports, penetrations, and welds joining the piping to adjacent components within the prescribed boundaries.

Descriptions of systems that contain ASME B31.1 piping and components including their functions are described in the system chapters.

##### **3.6.7.3.1 Overpressure Protection**

The details and certification of overpressure protection design for each piping system are in the System Overpressure Protection Reports.

##### **3.6.7.3.2 Boundaries**

The boundaries of the ASME B31.1 piping in each system are outlined in the respective system P&ID and indicated in the simplified diagrams provided in each chapter.



### **3.6.7.3.3 Classifications**

#### **Code Classification**

Detailed classifications of pipe and components are defined in the system design documents.

Portions of the ASME BPVC Section III Division 1 Class 2 or 3 piping system analysis may contain ASME B31.1 piping beyond a normally closed valve which may define the boundary out to the first anchor in the ASME B31.1 piping system.

The pipe supports attached to the ASME B31.1 piping meet the appropriate requirements of ASME B31.1.

#### **Seismic Classification**

Seismic categories are to be in accordance with those listed on the system design documents.

#### **Energy (High/Moderate) Classification**

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Subsection 3.4.4.2 for further explanation.

### **3.6.7.3.4 Materials**

#### **Material Specifications**

The material properties used in ASME B31.1 system analysis are in accordance with ASME B31.1.

#### **Examination and Repair**

The examination and repair of all ASME B31.1 piping materials and welds are performed using the methods and acceptance standards as specified in ASME B31.

The recommended practice for operation, maintenance, and modification of ASME B31.1 piping, and components is in accordance with the applicable local jurisdiction standard and code.

#### **Fracture Toughness Requirements**

The requirements of ASME B31T, *Standard Toughness Requirements for Piping*, Paragraphs 3, 4, and Appendix A are met.

### **3.6.7.3.5 Design Conditions**

#### **Design Service Life**

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years plus any additional time in-service for startup and decommissioning activities as applicable.

#### **Design Pressures and Temperatures**

The design pressures and temperatures of each piping system are identified in the respective system design documents.

#### **Design Duty Cycles**

Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

#### **Environmental Conditions**

Consideration of environmental conditions for functional qualification is not applicable to ASME B31.1 piping systems.

Recommended practices related to the protection of piping systems against detrimental effects of environmental conditions are provided in ASME B31.1 Appendices IV and V.

#### **3.6.7.3.6 Design Input Loads**

##### **Test Loads**

The only test loads on the piping system are due to hydrostatic testing.

##### **Static Loads**

###### **Pressure**

The design pressure and operating pressure for each system/component are as specified in the respective Process Flow Diagram.

###### **Weight**

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe is considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and refueling outage is designed to accommodate the increased weight.

###### **Thermal Expansion**

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses.

The installation temperature for the piping systems is defined as a temperature of 21° C for non-nuclear (ASME B31.1) piping.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design. Thermal anchor movements of less than 1.6 mm are considered negligible and do not need to be considered in the analysis.

###### **Thermal Attenuation/Stratification**

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

#### **3.6.7.3.7 Dynamic Loads**

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement). Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Turbine Stop Valve Closure
- Reactor Pressure Vessel and Containment Isolation Valve Transients

#### **3.6.7.3.8 *Plant Events and Load Combinations***

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

#### **Load Combinations**

The load combinations and acceptance criteria presented in this specification are applicable to all ASME B31.1 piping systems, structures, and components within the scope of this document.

#### **Load Combinations for Piping and Components**

The load combinations and acceptance criteria in Table 3.6-12 are applied to the analysis of ASME B31.1 piping systems and components.

### **3.6.8 Threaded Fasteners – Codes for ASME BPVC Section III Division 1 Class 1, 2, and 3**

#### **3.6.8.1 Material Selection**

Material used for threaded fasteners complies with the requirements of ASME BPVC Section III Division 1 Article NB-2000, NCD-2000, or NF-2000 as appropriate. Fracture toughness testing is performed in accordance with ASME BPVC Section III Division 1 Subarticle NB-2300, or NCD-2300, as appropriate. For verification of conformance to the applicable ASME BPVC requirements, a chemical analysis is required for each heat of material and testing for mechanical properties is required on samples representing each heat of material and, where applicable, each heat-treat lot.

The criteria of ASME BPVC Section III Division 1 Subarticle NB-2200, or NCD-2200, rather than the material specification criteria applicable to the mechanical testing is applied if there is a conflict between the two sets of criteria. For threaded fasteners, documentation related to fracture toughness (as applicable) and certified material test reports are provided as part of the ASME BPVC Section III Division 1 records that are provided at the time the parts are shipped and are part of the required records that are maintained at the site.

Threaded fasteners are selected for compatibility with the materials of the component being joined and the piping system fluids. The selection process considers deterioration that may occur during service as a result of corrosion, radiation effects, or instability of material.

#### **3.6.8.2 Special Materials Fabrication Processes and Special Controls**

The design of threaded fasteners complies with ASME BPVC Section III Division 1 Article NB-3000 or NCD-3000, as appropriate. Fabrication of threaded fasteners complies with ASME BPVC Section III Division 1 Article NB-4000, NCD-4000, as appropriate. Inspection of threaded fasteners complies with ASME BPVC Section III Division 1 NB-2500, or NCD-2500, as applicable.

#### **3.6.8.3 Pre-service and In-service Inspection Requirements**

Pre-service and In-service requirements of ASME BPVC Section III Division 1 Class 1, 2, and 3 Mechanical Systems and Components are based on a graded approach with SC1 equipment receiving the most pre-service required qualification. Chapter 3, Section 3.9 Equipment Qualification provides the required qualifications and tests for safety components. Chapter 3, Subsection 3.10.5 provides the In-service Inspection requirements for SSCs.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**3.6.9 References**

- 3.6-1 ASME BPVC-IID (Metric), "Section II - Materials - Part D - Properties - (Metric)," American Society of Mechanical Engineers.
- 3.6-2 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components - Appendices," American Society of Mechanical Engineers.
- 3.6-3 ASME BPVC-III NB, "Section III - Rules for Construction of Nuclear Facility Components, Subsection NB - Class 1 Components," American Society of Mechanical Engineers.
- 3.6-4 ASME BPVC-III NCD, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NCD - Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 3.6-5 ASME BPVC-III NCA, "Section III - Division 1 and 2 - Subsection NCA, Rules for Construction of Nuclear Facility Components - General Requirements for Division 1 and Division 2," American Society of Mechanical Engineers.
- 3.6-6 ASME BPVC-III NE, "Section III Division 1 - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NE - Class MC Components," American Society of Mechanical Engineers.
- 3.6-7 ASME BPVC-III NF, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NF - Supports," American Society of Mechanical Engineers.
- 3.6-8 ASME BPVC-III NG, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NG - Core Support Structures," American Society of Mechanical Engineers.
- 3.6-9 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components – Appendices - Mandatory Appendix II," American Society of Mechanical Engineers.
- 3.6-10 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 3.6-11 ASME B31T, "Standard Toughness Requirements for Piping," American Society of Mechanical Engineers.
- 3.6-12 ASME B31.3, "Process Piping," American Society of Mechanical Engineers.
- 3.6-13 CSA N289.4, "Testing procedures for seismic qualification of nuclear power plant structures, systems, and components." CSA Group.
- 3.6-14 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 3.6-15 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.6-16 CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, Version 1.
- 3.6-17 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.6-18 IEC/IEEE 60780-323, "Nuclear facilities – Electrical equipment important to safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.6-19 IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.6-20 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-1: Applicable Pressure Boundary Codes and Standards**

<b>Code or Standard Number</b>	<b>Title/Description</b>
ASME Section III Division 1 BPVC Section II	Materials
ASME BPVC Section III, Division 1	BPVC Section III, Rules for Construction of Nuclear Facility Components (NCA, NB, NCD, NE, NF, NG)
ASME BPVC Section V	Nondestructive Examination
ASME BPVC Section VIII, Division 1	BPVC Section VIII-Rules for Construction of Pressure Vessel
ASME BPVC Section IX	Welding and Brazing Qualifications
ASME BPVC Section XI	Rules for In-service Inspection of Nuclear Power Plant Components
ASME B31.1	Power Piping
ASME B31.3	Process Piping
ASME B31.5	Refrigeration Piping and Heat Transfer Component Code
ASTM	American Society for Testing and Materials (various material and forms specifications for piping and related components)
API-620 (or equivalent)	Design and Construction of Large, Welded, Low-Pressure Storage Tanks
API-650 (or equivalent)	Welded Tanks for Oil Storage
AWWA-D100	Welded Carbon Steel Tanks for Water Storage

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-2: Load Combinations and Acceptance Criteria**

<b>Plant Event / Event Combination</b>	<b>Service Loading Combination<sup>(1)(2)(3)(10)</sup></b>	<b>Comments</b>	<b>ASME Service Level<sup>(4)</sup></b>
Design	$P_D + T_D + R_D$ Design		N/A
Normal Operation	N		A
Plant/System AOO	(a) $N + AOO_A$ (b) $N + AOO_B$		B
Normal Operation + SOE	$N + SOE^{(11)}$	OPG/CSA requirement for SOE <sup>(11)</sup> for Level B	B <sup>(6) (7)</sup>
Design Basis Accident	(a) $N + DBA_A$ (b) $N + DBA_B$ Loadings	OPG/CSA requirement for DBE <sup>(11)</sup> for Level C	C
Design Extension Condition	(a) $N^{(5)} + DEC_A$ (b) $N^{(5)} + DEC_B$	OPG/CSA requirement for CLE <sup>(11)</sup> for Level D	D
Test <sup>(9)</sup>	$P_t + T_t + D_t$		Testing Limit <sup>(8)</sup>

- (1) The service loading combination also applies to Seismic Category A and B instrumentation and electrical equipment.
- (2) For vessels, loads induced by the attached piping are included as identified in their design specification. For piping systems, water (steam) hammer loads are included as identified in their design specification.
- (3) The method of combination of the loads is in accordance with NUREG-0484, Revision 1.
- (4) Service level requirements are only applicable to ASME BPVC Code, Section III components. The service levels are as defined in appropriate subsection of ASME BPVC Code, Section III, Division 1.
- (5) The Reactor Coolant Pressure Boundary (RCPB) is evaluated in the load combination using the maximum pressure expected to occur during the Postulated Accident.
- (6) Applies only to fatigue evaluation of ASME BPVC Code Class 1 components and core support structures.
- (7) For ASME BPVC Code Class 1, 2 and 3 piping changes and additions to ASME BPVC Code Section III NB-3600, NCD-3600 may be necessary to evaluate and meet stress limits.
- (8) Testing limits are per ASME BPVC Code Section III NB-3226.
- (9) Test conditions are only applicable to ASME components.
- (10) Nomenclature:
  - a.  $AOO_x$  Loads for AOO event x
  - b. D Dead Load
  - c.  $D_t$  Dead Load for Test Condition
  - d. DBE Design Basis Earthquake Loads
  - e.  $DEC_x$  Loads for DEC event x
  - f. N Normal Operation Loads
  - g.  $P_D$  Design Pressure
  - h.  $P_t$  Test Pressure
  - i.  $DBA_x$  Loads for DBA event x

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- j.  $R_D$  Design Mechanical Loads
  - k.  $R_t$  Test Mechanical Loads
  - l.  $T_D$  Design Temperature
  - m.  $T_t$  Test Temperature
- (11) For. OPG, SOE, DBE and CLE are the earthquake levels defined in Section 3.2.5. Per OPG PSAR,  $SOE = (1/3) * DBE$ . CLE is defined in Supporting documents (6), but is expected to be  $(1.5 \text{ to } 1.67) * DBE$ .



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-3: Comparison of Event Frequency to Plant Conditions and Service Loadings**

<b>Design Condition (DC)</b>	<b>ASME Service Level</b>	<b>Quantitative Frequency (F) (1/year)</b>
Normal Planned Operation (DC-1)	A; - loading during plant startup, operation, refueling, and shutdown.	Planned Operation
Anticipated Operational Occurrences (AOO) (DC-2)	B; - incidents of moderate frequency occasional, infrequent loadings without sustaining any damage or reduction in function.	$< 1\text{E-}02$
Design Basis Accidents (DBAs) (DC-3)	C; - incidents of low frequency – infrequent loadings causing no significant loss of integrity.	$1\text{E-}02 > F \geq 1\text{E-}05$
Design Extension Conditions (DECs) (DC-4)	D; - incidents of extremely low frequency loadings associated with beyond design basis accidents.	$F \leq 1\text{E-}05$

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-4: Normal Operating Events (DC-1)**

<b>Description</b>	<b>Number of Cycles/60 Years</b>
Boltup	72
Startup	200
Turbine Roll and Increase to Rated Power	200
Daily/Weekly Load Reduction and Recovery	20,805
Rod Sequence/Pattern Change	30
Rated Power Operation	-
Reduction to 0% Power	200
Hot Standby	200
Shutdown	200
Vessel Flooding/Shutdown Cooling	72
Unbolt	72
Refuel	72

**Table 3.6-5: Test Events (DC-1)**

<b>Description</b>	<b>Number of Cycles/60 Years</b>
Design/System Leakage Hydrostatic Testing	150
Turbine Stop Valve Test	3,120
Turbine Bypass Valve Test	720
Turbine control Valve Test	720
MSIV Closure Test	720

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-6: Anticipated Operational Occurrences (DC-2)**

<b>Description</b>	<b>Number of Cycles/60 Years</b>
Loss of Feedwater Heaters – Partial	50
Loss of Feedwater Heaters – Total	10
Rod Withdraw Error at Startup	7
Turbine Generator Trip. Load Rejection – with Bypass	60
Turbine Control Valve Fail Open	1
Loss of Feedwater	15
Loss-of-Offsite Power	8
Loss of Condenser Vacuum	10
Inadvertent MSIV Closure (all MSIVs)	20

**Table 3.6-7: Design Basis Accidents (DC-3)**

Description	Number of Cycles/60 Years
Improper Startup – Hot Cleanup Water System	1 (freq $\leq$ 0.1)
Turbine Generator Trip. Load Rejection – Without Bypass	1 (freq $\leq$ 0.1)
Reactor Overpressure – Backup Scram	1 (freq $\leq$ 0.1)
Shutdown due to Inadvertent Isolation Condenser System (ICS) Initiation	1 (freq $\leq$ 0.1)
Inadvertent Sodium Pentaborate Injection	1 (freq $\leq$ 0.1)
Excessive Cooldown Rate	2 (freq $\leq$ 0.1)

**Table 3.6-8: Design Extension Condition (DC-4)**

<b>Description</b>	<b>Number of Cycles/60 Years</b>
Bounding Transient without Scram	$\leq 0.001$
Pipe Rupture – Loss-of-Coolant Accident	$\leq 0.001$
Ultimate Overpressure Protection	$\leq 0.001$

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-9: Summary of Cycles of Events**

<b>Event #</b>	<b>Description</b>	<b>Design Basis Number of Cycles</b>
1	Boltup	72
2	Design/System Leakage Hydrostatic Testing	150
3	Startup	200
4	Turbine Roll and Increase to Rated Power	200
5/6	Daily/Weekly Load Reduction and Recovery	20,805
7	Rod Sequence/Pattern Change	30
8	Loss of Feedwater Heaters – Partial	50
9	Loss of Feedwater Heaters – Total	10
10/11	Turbine Generator Trip, Other Scrams with Bypass Flow	67
12	Rated Power Operation	-
13	Reduction to 0% Power	200
14	Hot Standby	200
15	Shutdown	200
16/17	Vessel Flooding/Shutdown Cooling	72
18	Unbolt	72
19	Refuel	72
20	Scrams Without Bypass	55
21	Improper Startup – Hot Reactor Water Cleanup System	1
22	Reactor Overpressure – Backup Scram	1
23	Shutdown due to Inadvertent Isolation Condenser System (ICS) Initiation	1
24	Improper Startup/Sodium Pentaborate Injection	1
25	Excessive Cooldown Rate	2
26	Bounding Transient Without Scram	1
27	Pipe Rupture – Loss-of-Coolant Accident	1
28	Ultimate Overpressure Protection	1

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-10: Load Combinations and Acceptance Criteria for ASME  
BPVC Section III Division 1 Class 1 Piping Systems**

Condition	Load Combination for all Terms <sup>(2)(3)</sup>	Acceptance Criteria per ASME Code <sup>(1)(4)</sup>
Design	PD + WT	NB-3652
Service Level A and B <sup>(5)</sup>	PP, TE, $\Delta T1$ , $\Delta T2$ , TA-TB, AOO, DBEI, DBED	NB-3653
Service Level B	PP + WT + AOO	NB-3654
Service Level C	PP + WT + DBA Where DBA includes but is not limited to: LOCA DBE	NB-3655
Service Level D	PP + WT + DEC Where DEC includes but is not limited to: SRSS (DBE+LOCA)	NB-3656

(1) Fatigue usage and stress limits are reduced for piping locations exempt from pipe break consideration.

(2) Where:

- a. WT = Dead Weight
- b. PD = Design Pressure
- c. PP = Peak Pressure or the Operating Pressure Associated with that transient
- d. DBEI = Design Basis Earthquake (inertia Effect)
- e. DBED = Design Basis Earthquake (Anchor Displacement Loads)
- f. DBE = Design Basis Earthquake includes both DBEI and DBED which are combined using SRSS method
- g. AOO = Anticipated Operational Occurrence
- h. DBA = Design Basis Accident
- i. DEC = Design Extension Condition

(3) LOCA is intended to represent loads and the appropriate combination of loads resulting from postulated line breaks including but not limited to Acoustic Inertial, Jet Reaction, and Jet Impingement loads

(4) ASME BPVC SECTION III NB-2021

(5) DBEI and DBED are Service Level C loads but must be considered for fatigue usage.



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.6-11: Load Combinations and Acceptance Criteria for ASME BPVC III Division 1  
Class 2 and 3 Piping Systems**

<b>Service Level</b>	<b>Load Combination for all Terms<sup>(1)(2)(3)</sup></b>	<b>Acceptance Criteria per ASME Code<sup>(4)(5)</sup></b>
Design	PD + WT	NCD-3652
A & B	TE	NCD-3653.2
A & B	Single Non-repeated Anchor Movement	NCD-3653.2
A & B	PD + WT + TE	NCD-3653.2
B	PP + WT + AOO Where AOO includes but is not limited to: TSV	NCD-3653.1
C	PP + WT + DBA Where DBA includes but is not limited to: LOCA DBE	NCD-3654.2
C	PP	NCD-3654.1
D	PP + WT + DEC Where DEC includes but is not limited to: SRSS (DBE + TSV) SRSS (DBE + LOCA)	NCD-3655
D	PP	NCD-3655

(1) TSV loads are used for MS lines only

(2) Where:

- a. WT = Dead Weight
- b. PD = Design Pressure
- c. PP = Peak Pressure or the Operating Pressure Associated with that transient
- d. DBEI = Design Basis Earthquake (inertia Effect)
- e. DBED = Design Basis Earthquake (Anchor Displacement Loads)
- f. DBE = Design Basis Earthquake includes both DBEI and DBED which are combined using SRSS method
- g. AOO = Anticipated Operational Occurrence
- h. DBA = Design Basis Accident
- i. DEC = Design Extension Condition

(3) LOCA is intended to represent loads and the appropriate combination of loads resulting from postulated line breaks including but not limited to Acoustic Inertial (ACI), JR, and JI loads

(4) ASME BPVC SECTION III NCD-2021

(5) Stress limits are reduced for piping locations exempt from pipe break consideration.

**Table 3.6-12: Load Combinations and Acceptance Criteria for Non-Safety Class Power Piping Systems**

Description	Load Combination	Acceptance Criteria per ASME Code <sup>(2)</sup>
Sustained	Design Pressure + Weight + other Sustained Loads	Paragraphs 102.3 and 104.8.1
Occasional	Design Pressure + Weight + Other Sustained Loads + Seismic	Paragraphs 102.3 and 104.8.2
Occasional	Design Pressure + Weight + Occasional event other than Seismic	Paragraphs 102.3 and 104.8.2
Thermal	Displacement Load Ranges	Paragraphs 102.3 and 104.8.3
Test	Test Pressure + Weight	Paragraph 102.3.3

(1) Stated in CSA N289.3: Clause 7.5.1 (Reference 3.6-15). For Class 6 piping in accordance with ASME B31.1-2020 rules, the k factor in the equation for stresses due to occasional loads including seismic loading is increased to 1.8. Alternatively, a conservative approach can be adopted in which the seismic stresses in the stress combination for occasional loads can be multiplied by factor 2/3 with the k factor equal to 1.2.

(2) ASME B31.1-2020

### **3.7 General Design Aspects for Instrumentation and Control Systems and Components**

The BWRX-300 Distributed Control and Information System (DCIS) is an integrated control and monitoring system for the power plant. The DCIS is arranged in three safety classified DCIS segments that have appropriate levels of hardware and software quality corresponding to the system functions they control and their allocation to the Defense Lines (DL). The DCIS provides control, monitoring, alarming and recording functions. Although normally integrated, the various components of the DCIS are designed to operate independently.

The relationship between Instrumentation and Control (I&C) Functions and plant-level DLs is described in Chapter 7, Section 7.1.1. The classification of I&C systems is described in Chapter 7, Section 7.1.2, and is based on the general classification criteria described in Sections 3.2.1 and 3.2.2. The I&C system of systems is described in Chapter 7, Section 7.2. The individual I&C systems are described in Chapter 7, Section 7.3.

#### **3.7.1 Performance**

The system design bases, and associated safety functions, are described for the DL3 systems in Chapter 7, Subsection 7.3.1.2, for the DL4a systems in Subsection 7.3.2.2, for the DL2 systems in Subsection 7.3.3.2, and for the non-classified systems in Subsection 7.3.4.2.

#### **3.7.2 Design for Reliability**

The system reliability requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

#### **3.7.3 Independence**

The system independence requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.3, for the DL4a systems in Subsection 7.3.2.3.3, for the DL2 systems in Subsection 7.3.3.3.3, and for the non-classified systems in Subsection 7.3.4.3.3.

#### **3.7.4 Qualification**

The system qualification requirements are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.1, for the DL4a systems in Subsection 7.3.2.3.1, for the DL2 systems in Subsection 7.3.3.3.1, and for non-classified systems in Subsection 7.3.4.3.1.

#### **3.7.5 Verification and Validation**

The system verification and validation requirements for I&C systems are described in Chapter 7, Section 7.4.3.

#### **3.7.6 Failure Modes**

The application of the single failure criterion to DL3 systems is described in Chapter 7, Subsection 7.3.1.3.3. The effects of failures and associated design features to minimize or eliminate adverse effects of anticipated failures are described for the DL4a systems in Subsection 7.3.2.3.3, for the DL2 systems in Subsection 7.3.3.3.3, and for the non-classified systems in Subsection 7.3.4.3.3.

The use of diversity to eliminate common cause failure vulnerabilities or minimize the effects of postulated common cause failures is described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.5, for the DL4a systems in Subsection 7.3.2.3.5, for the DL2 systems in Subsection 7.3.3.3.5, and for the non-classified systems in Subsection 7.3.4.3.5.

### **3.7.7 Control of Access to Equipment**

The system security requirements (including control of access) are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.4, for the DL4a systems in Subsection 7.3.2.3.4, for the DL2 systems in Subsection 7.3.3.3.4, and for the non-classified systems in Subsection 7.3.4.3.4.

### **3.7.8 Quality**

The codes and standards used for the I&C systems are described in Chapter 7, Section 7.1.3. The system quality requirements are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.1, for the DL4a systems in Subsection 7.3.2.3.1, for the DL2 systems in Subsection 7.3.3.3.1, and for the non-classified systems in Subsection 7.3.4.3.1.

### **3.7.9 Testing and Testability**

The system testing requirements (including design features to support testability) are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

### **3.7.10 Maintainability**

The system maintainability requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

### **3.7.11 Identification of Items Important to Safety**

The I&C system classification information is described in Section 7.1.2.

### **3.8 General Design Aspects for Electrical Systems and Components**

The BWRX-300 electrical power system has been designed as a minimum to meet the requirements of CNSC REGDOC 1.1.2 and CNSC REGDOC 2.5.2.

The electrical power system design is a 60 Hz Alternating Current (AC) power system, with 4.16 kV for the Medium Voltage (MV) level and 600 V for the Low Voltage (LV) level.

The off-site electrical system is provided and managed by OPG. The function of the BWRX-300 off-site electrical system is to provide electrical power to the Hydro One managed grid that is compatible and consistent for OPG purposes. The output of the BWRX-300 is monitored for over voltage and over/under current as protective design features to prevent possible grid disruptions. The off-site power system can be automatically or manually disconnected from the grid if the electrical power is found to be disrupted for any reason.

On-site electrical systems are designed to support the normal operations of the BWRX-300. A unique feature of the BWRX-300 plant is that the on-site AC power system is not required to be operational to support the safe shutdown of the reactor and for at least the first 72 hours following shutdown. The reactor cooldown is accomplished through natural circulation and passive cooling via the ICS system.

The off-site preferred power system is designed to provide a continuous source of power to the on-site AC power system throughout plant startup, normal operation (including shutdown), and abnormal operations. The off-site power system provides no credited safety function. As a result, the total loss-of-offsite power results in no impact on nuclear safety.

Refer to Chapter 8 – Electrical Power for a detailed discussion on the Electrical power systems for the BWRX-300.

The on-site AC power system consists of SCN, SC1, SC2, and SC3 power systems. The two off-site power sources provide the normal preferred and alternate preferred AC power to SCN, SC1, SC2 and SC3 loads.

The normal preferred off-site power source is connected to the GSU, which is connected to the plant generator and the UAT. The normal preferred power source is distributed from the UAT secondary windings to MV SCN busses, which further distribute the power to SCN loads and the SC3 LV busses. The SC3 LV busses serve LV SC3 loads and provide normal AC power to the SC1 and SC2 electrical power systems.

The alternate preferred off-site power source is connected to the RAT, which has two MV secondary windings like the UAT. The RAT provides alternate power feeds to the MV SCN busses for cases when the UAT is not in-service.

The SC3 LV busses also have backup power in the form of standby diesel generators. Each SC3 LV bus is connected to a standby diesel generator that automatically starts and loads if the normal power to the SC3 LV bus becomes unavailable (loss of power or degraded).

There are three divisions of SC1 DC power, two load groups of SC2 DC power, and 2 sets of SCN DC power connected to the diesel-backed SC3 busses. Add that each DC power system includes battery chargers, batteries, and UPSs to supply uninterruptible AC and DC power during loss of power events.

The BWRX-300 electrical AC power systems (on-site or off-site) are not relied upon to support the safe shutdown and cooldown of the reactor in the event of a design basis accident. No operator actions are credited in the safe shutdown or cooldown of the reactor in the event of a design basis accident.

### **3.8.1 Redundancy**

As discussed above, two off-site power sources provide the normal preferred and alternate preferred AC power to SCN, SC1, SC2 and SC3 loads. In the event of total loss-of-offsite power sources SC3 standby diesel generators are provided to power the plant SC1, SC2 and SC3 loads.

Three divisions of SC1 DC power are not only redundant to each other, but also have redundant UPSs in each divisions for further reliability. The SC2 DC power load groups are redundant to each other as well. There are also two sets of SCN DC systems that can provide redundant power to select equipment as needed.

There are two redundant SC2 Direct Current (DC) load groups and one SC3 Direct Current (DC) load group each with a UPS to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 Direct Current (DC) divisions with UPS to provide power to SC1 loads.

Redundancy for the BWRX-300 electrical power systems is discussed in more detail in Chapter 8.

### **3.8.2 Independence**

As discussed above, in the event of total loss-of-offsite power sources two on-site SC3 standby diesel generators are provided to power the plant SC1, SC2 and SC3 loads. Either SDG can support the required SC1, SC2, and SC3 loads needed for active decay heat removal. The SDG's are located in independent fire-barriered rooms.

The 3 divisions of SC1 DC power are electrically and physically independent from each other. There are no electrical connections between the divisions and the equipment is located by division in separate fire and flood-barriered rooms.

It is also the same for the SC2 load groups, (i.e., the two SC2 load groups are similarly independent from each other).

There are two independent SC2 Direct Current (DC) load groups and one SC3 Direct Current (DC) load group each with a UPS to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 Direct Current (DC) divisions with UPS to provide power to SC1 loads.

Independence of the electrical power systems and components is discussed in more detail in various Chapter 8 sections. Refer to Chapter 8 for further discussion of this topic.

### **3.8.3 Diversity**

The EDS is designed along a Defence-in-Depth philosophy and along Defense Lines. Section 3.6 provides a discussion on philosophy. The electrical systems are diverse from each based on defense lines.

### **3.8.4 Controls and Monitoring**

On-site and Off-site electrical power system controls and monitoring for the BWRX-300 will be accomplished by both Main and Secondary Control rooms monitors or controls that are remote "at the panel" monitoring and controls should it be necessary to operate the electrical systems in a remote "away from the CR" fashion.

Controls and Monitoring is discussed in Chapter 8.

### **3.8.5 Identification**

Refer to Section 8.4 for details on the electrical system safety classification and a description of the major electrical power system equipment.

### **3.8.6 Capacity and Capability of Systems for Different Plant States**

The capacity and capability of the Electrical Power Systems is designed to provide a minimum of 100% of the required electrical loading needed for the normal operation of the BWRX-300. Equipment sizing includes consideration of design margin as appropriate for all facets of plant operation.

As stated above, the BWRX-300 does not rely on electrical power to safely shutdown and cool the reactor. Electrical power is not relied upon to place the reactor into a safe shutdown and to maintain the reactor in a safe shutdown condition.

As mentioned previously, SDG capacity can support required SC1/2/3 loads needed for active decay heat removal.

DC power from batteries will be used primarily to monitor the cooldown and condition of the reactor.

The capacity and capability of electrical power system is further discussed in Chapter 8.

### **3.8.7 External Grid and Related Issues**

External Grid operation and management is the responsibility of OPG. The BWRX-300 safety design does not require off-site power to be present to mitigate any design basis accidents.

OPG's grid connection project is currently in the conceptual and planning stage.

With input and interfacing support, OPG plans on designing and building a local switchyard to consolidate power output from the BWRX-300 SMR Facility and connect it with Ontario electrical power grid. Hydro One is the grid transmitter and the Independent Electricity System Operator (IESO) is the electrical system operator.

At this time, OPG is expected to be the operator of the local switchyard via the Main Control Room (MCR) in the SMR Facility. The SMR Facility electrical AC power system will have two high voltage connections with the local switchyard at a 230kV voltage level. One line to output power from the Generator Step Up Transformer (GSU) and one line to supply power to the Reserve Auxiliary Transformer (RAT). The local switchyard will have two redundant 230kV connections with the transmitter. Each line will be designed to transmit the full generation capacity of the SMR Facility. The transmitter is responsible for building the transmission infrastructure needed to connect the local switchyard to Clarington TS, 22km North of the DNNP site. The two lines are expected to share the same tower structure. *(The 230kV voltage level and connection with Clarington TS is to be confirmed in 2022 through an IESO Feasibility Study.)*

The local switchyard will be of an indoor Gas Insulated Switchgear type, following a breaker and half arrangement with two redundant busses. The local switchyard will be designed to have local and remote-control capability. The plan for the local DNNP switchyard is that it will be located North of the SMR Facility, East of the Extended Holt Rd and South of the CN Rail tracks. The local switchyard control and protection designs will be coordinated with the SMR Facility controls and protections to meet IESO, NPCC and NERC codes and standards.

**Power Quality**

The BWRX-300 electrical power systems will be monitored for power quality issues (voltage/frequency/harmonics) that may arise and maintained such that any abnormal fluctuations in the voltage, current or capacity is alarmed in the Main Control Room so operators can evaluate and manually respond to the alarm condition.



### **3.9 Equipment Qualification**

#### **3.9.1 Purpose**

This section defines the requirements related to equipment qualification in alignment with CNSC REGDOC-2.5.2, Section 5.5 (Reference 3.9-1).

Equipment qualification is the process carried out (including the generation and maintenance of evidence) to ensure SSC can perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

The conditions impacting equipment qualification include seismic/dynamic, environmental, functional/aging stressors, and electromagnetic interference.

#### **3.9.2 Scope**

Equipment qualification requirements are applied to BWRX-300 equipment based on the assigned safety classification and seismic categorization of SSC (as described in Section 3.2), and to certain post-accident monitoring equipment.

Equipment qualification considers all normal operating conditions in which the SSC are expected to operate including conditions arising from maintenance and testing, and also, the conditions arising from AOOs, DBAs, and internal and external hazards.

While DEC conditions are generally considered outside of the scope of a qualification program, guidance is provided for demonstrating with reasonable assurance that equipment credited to perform under DEC conditions will survive to perform its function. See Subsection 3.9.3.5 for consideration of a Beyond-Design Basis Earthquake (BDBE) and Subsection 3.9.4.1 for Environmental Qualification considerations.

The focus of this section is on qualification of mechanical and electrical equipment. Mechanical equipment consists of items of a facility including pumps, valves, vessels, and piping whose function is required to ensure the safe operation or safe shutdown. Electrical equipment consists of all electrical power and Instrumentation and Control (I&C) equipment, which includes all analog (non-digital) and digital I&C components. Computer-based I&C equipment is a subset of digital I&C components.

Qualification of civil structures is covered in Section 3.3.

#### **3.9.3 Seismic**

##### **3.9.3.1 General**

Seismic qualification is a subset of equipment qualification that is the verification, through testing, analysis, or other methods, of the ability of an SSC to perform its intended function during and/or following a designated earthquake. The dynamic loads of Reactor Building Vibrations (RBVs) and events caused by hydrodynamic loads are also considered. Seismic and dynamic qualification of BWRX-300 equipment and associated supports meets the requirements and recommendations of the CSA N289 series (References 3.9-2 To 3.9-6) as endorsed by CNSC REGDOC-2.5.2 (References 3.9-1), and IEC/IEEE 60980-344 (Reference 3.9-7).

The requirement for seismic qualification is based on the seismic categorization of SSC and the earthquake level they are required to withstand during and/or after the seismic event. Seismic categorization of BWRX-300 SSC is described in Section 3.2. Seismic Category A and Seismic Category B SSC are most important and have the most stringent requirements for functional integrity during and following a seismic event. Per regulatory guidance of CNSC REGDOC-2.5.2, Section 5.13.1 (Reference 3.9-1), SSC that are classified as Seismic Category A and Seismic

Category B are seismically qualified to withstand the effects of a DBE. The site-specific DBE is defined in Subsection 3.3.1.

BWRX-300 equipment Seismic Categories are identified in Appendix 3A Table 3.12-1. Seismic Categorization of Structures is provided in Section 3.3, Table 3.3-1.

### **3.9.3.2 Methods for Seismic Qualification**

Seismic and dynamic qualification of equipment and associated supports are accomplished by test, analysis, or a combination of testing and analysis. Seismic and dynamic qualification of equipment and associated supports designated as SC1 is accomplished by testing. Seismic and dynamic qualification of equipment and associated supports designated as SC2 may be accomplished by analysis or a combination of testing and analysis.

Qualification by actual seismic experience (also referred to as seismic qualification by similarity), as described in IEC/IEEE 60980-344 (Reference 3.9-7) and CSA N289.1 (Reference 3.9-2), is also utilized as appropriate considering the limitations identified in CSA N289.1, Annex D.3 (Reference 3.9-2).

The selection of qualification method to be used is largely a matter of engineering judgment for cases where testing is not required. When both test and analysis are defined as acceptable methods, the deciding factors considered (as applicable) for choosing between tests or analysis include magnitude and frequency of seismic and RBV dynamic loadings, environmental conditions associated with the dynamic loadings, nature of the safety category function(s), size and complexity of the equipment, dynamic characteristics of expected failure modes (structural or functional), and partial test data upon which to base the analysis.

Tests or analyses of assemblies are preferable to tests or analyses on separate components (e.g., a motor and a pump, including the coupling and other appurtenances, should be tested or analyzed as an assembly), unless deemed not practical. Equipment that has been previously qualified by means of tests and analyses equivalent to those required for the current qualification program are used if proper documentation of such tests and analyses is available.

For equipment defined as requiring test for qualification, analysis by similarity may be used if similar equipment is being or has been qualified by test.

#### **3.9.3.2.1 Testing**

Testing of BWRX-300 SSC for seismic qualification is conducted in accordance with CSA N289.4 (Reference 3.9-5) IEC/IEEE 60980-344 (Reference 3.9-7).

Seismic qualification by testing is typically used for SSC that will be performing an active function and are required to change state during or following a seismic event to perform a safety category function, while maintaining structural and/or pressure boundary integrity. Seismic testing can identify contact chatter or unauthorized change of state of contact in electrical and I&C components during seismic excitation.

The dynamic test sequence includes as applicable, vibration conditioning, exploratory resonance search, low-level earthquake loading (one-half DBE) including Reactor Building Vibrations (RBV) dynamic loads and the DBE loading including RBV dynamic loads.

Dynamic tests are performed with the equipment subjected to nominal operating service conditions. Significant, normal operating loads such as electrical, mechanical, pressure, and thermal are included. Where normal operating loads cannot be included in the dynamic tests, supplemental analysis is used to qualify the equipment for those effects. If there is any dynamic coupling due to interacting equipment, it is considered.

For equipment located in multiple locations, the enveloping upper bound seismic condition limits are used to eliminate the need for multiple qualification tests, unless otherwise specified.

### **Resonance Tests**

When required, exploratory resonance search tests (such as sine sweeps or random vibration) are used for equipment to help determine the method of test or analysis that would be best for qualification and/or determine the dynamic characteristics such as the resonance frequencies of the equipment, mode shapes and damping values.

Sine sweep resonance search is the preferred method and is performed by running a continuous sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude at which resonance can be determined.

Resonance searches may be performed prior to and after the seismic test to determine any shifts in frequency caused by testing.

If resonance frequencies are present, the transmissibility between the input and the location of the equipment is determined by measuring the accelerations at the equipment location and calculating the magnification between it and the input.

Floor-mounted frequency testing can be used as another method to determine the resonance or natural frequencies for equipment.

### **Seismic Input Motion**

Dynamic load conditions are simulated by testing, using independent, random multi-frequency input or single frequency input motion (within equipment capability) over the frequency range of interest.

Acceptable justification for use of single frequency input includes, but is not limited to:

1. The characteristics of the required input motion are dominated by one frequency.
2. The anticipated response of the equipment is adequately represented by one mode.
3. The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelop the corresponding response spectra of the individual modes.
4. The time phasing of the inputs in the vertical or horizontal directions will be such that a purely rectilinear resultant input is avoided.

The actual input motion used during testing, for both multi and single frequency, envelops the applicable input motion (floor, wall, response, etc.) at the location(s) of the equipment under test.

When the equipment is qualified by dynamic test, the In-Structure Response Spectra (ISRS) or time histories, developed from the results of Soil-Structure Interaction (SSI) analyses as described in Section 3.3.1.2.7, representing the in-structure seismic response of the attachment point is used in determining required response spectra of input motion used for the test.

For the case of equipment having multiple supports with different dynamic motions, the effects of the multiple support attachment points must be considered in the dynamic qualification and can be accounted for by selecting an upper bound envelope of all the individual response spectra for these locations to calculate the maximum internal responses applicable to the equipment, unless otherwise specified.

Past testing demonstrates that Seismic Category A electrical equipment has critical damping ratios equal to or less than 5%. Hence, the required response spectra at 5% or less critical damping ratio are developed as input to the equipment base, unless identified otherwise.

### **Seismic Test**

The preferred test method for seismic qualification is shake table testing. Seismic testing is performed in a manner that demonstrates dynamic response characteristics and acceptability of the test specimen to withstand and maintain its function as required during the expected level of shaking. Test requirements are normally specified in the form of required response spectra at a specified damping value and confirmed by a Test Response Spectra (TRS) generated from the table motion.

The seismic test for DBE produces a TRS that envelops the applicable portion of the required response spectra as defined in the test specification (typically by a factor of 1.1) per CSA N289.4 (Reference 3.9-5). The approach is to apply 10% to the acceleration of the ISRS, developed from the results of SSI analyses as described in Section 3.3.1.2.6, which meets the recommendations of IEC/IEEE 60780-323 (Reference 3.9-8).

Testing for low-level earthquake loading and RBV dynamic loads is performed to demonstrate that the low-level earthquake loads combined with RBV dynamic loads do not degrade the continued structural and functional integrity of the equipment.

Testing for DBE loading and RBV dynamic loads are performed to demonstrate that equipment would perform its intended function(s) through DBE combined with RBV dynamic loads.

For both low-level earthquake and DBE seismic test runs, the input excitation TRS is required to envelop the specified required response spectra levels in accordance with CSA N289.4 (Reference 3.9-5) and Section 9 of IEC/IEEE 60980-344, (Reference 3.9-7).

If the TRS do not meet the requirements (i.e., do not envelop the required response spectra, do not demonstrate stationarity, do not demonstrate statistical independence) for the seismic test run, the test run is documented as unacceptable, adjustments may be required, and then the test is repeated.

Alternatively, per Clause 5.1.2.2.4 of CSA N289.4 (Reference 3.9-5), for acceptance in cases where TRS does not envelop required response spectra, the following criteria are applied:

- The number of points below the required response spectra shall not exceed 5
- The points shall not fall below the required response spectra by more than 10%
- Any two points below the required response spectra shall be at least 1 octave apart
- The points adjacent to the points that fall below the required response spectra shall be at least 10% above the required response spectra

For equipment that is subjected to vibration in its in-service condition, vibrational aging to its end of life condition must be completed prior to seismic testing (both low-level earthquake and DBE load tests).

For seismic qualification, the seismic input consists of five one-half DBE amplitude events (low-level earthquakes) followed by one DBE event. Alternatively, in accordance with Annex E of IEC/IEEE 60980-344 (Reference 3.9-7), a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half DBE events may be used followed by one full DBE event; however, in this case the amplitude shall not be below the minimum of one-half the DBE input motion.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

The preferred method for seismic testing is to use triaxial, multi-frequency testing. However, if justified, biaxial and single-axis testing is acceptable.

Multi-frequency, multi-axis dynamic tests (triaxial or biaxial) are used to qualify equipment with a single resonance or multiple resonances within the frequency range of interest or if the critical resonance frequencies cannot be ascertained.

Single frequency testing is allowed if:

1. It can be demonstrated that the component is subjected to no resonances, or one predominant resonance frequency that is not in the frequency range of interest, or if the resonance frequencies are widely separated and do not interact to reduce the fragility level in the frequency range of interest, or if otherwise justified.
2. Single-axis tests can only be used if the tests are designed to conservatively reflect the dynamic event at the equipment mounting locations or if the equipment being tested can be shown to respond independently in each of the three orthogonal axes or otherwise withstand the dynamic event at its mounting location.

Equipment is tested in a functionally operable condition to allow for the monitoring of safety requirements throughout the seismic testing.

Equipment is operated at appropriate times (as necessary) to demonstrate the ability to perform its safety category function throughout the seismic testing.

For Seismic Category A and B mechanical and electrical equipment, it is defined if the equipment must perform its safety category function before, during, and after seismic events (typical for most equipment), or only before and after seismic events (applicable to some equipment such as plant status display equipment).

The equipment damping value used for dynamic qualification is established in accordance with Section 5 of IEC/IEEE 60980-344 (Reference 3.9-7).

Documentation of seismic testing is in accordance with Section 13 of IEC/IEEE 60980-344, (Reference 3.9-7) and include, at minimum, locations of accelerometers, any existing resonance frequency(s) and transmission ratios, equipment damping coefficients if there is resonance over the range of the test response spectra, test equipment used, any modifications made to test specimen, hardware interface requirements, test methods, approval signature and dates, description of test facility, summary of results, equipment seismic qualification conclusions (including RBV dynamic loads), anomalies and their resolution, test data, and justification for using single-axis or single frequency tests for all items that are tested in this manner.

#### **3.9.3.2.2 Selection of Test Specimen**

Test specimens are selected as representative samples of the production equipment and supports that are covered by the qualification program. Test specimens are manufactured using the same process that are implemented for the production units. Variations in the configuration of the equipment are analyzed with supporting test data. For example, these variations may include mass distributions that differ from one cabinet to another. From test or analysis, it is determined which mass distribution results in the maximum acceleration or frequency content, and this worst-case configuration is used as the test specimen. The test report includes a justification that this configuration envelopes all other equipment configurations.

#### **3.9.3.3 Seismic Analysis**

Dynamic analysis or an equivalent static analysis is employed to qualify the equipment when analysis is chosen as the method for qualification per CSA N289.3, Section 6 (Reference 3.9-4).

The decision on using dynamic versus static analysis is typically defined based on whether the equipment is rigid or flexible.

If the fundamental frequency of the equipment is above the input excitation frequency (cutoff frequency of required response spectra) the equipment is considered rigid.

The search for the natural frequency is done analytically, if the equipment shape can be defined mathematically, or by prototype testing.

If the equipment is determined to be a rigid body (i.e., shown to have no resonance frequency within the expected frequency range) the static analysis method is able to be applied in place of dynamic analysis.

If the equipment is determined to be flexible (i.e., with the fundamental frequency of the equipment within frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied, unless justified otherwise.

If it is determined that either dynamic or static analysis can be used, in general, the choice of the analysis is based on the expected design margin, as the static coefficient method is more conservative than the dynamic analysis method.

For static analysis, the dynamic forces on each component can be obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment.

A static coefficient analysis may also be used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the required response spectra at the equipment mounting location, at a conservative and justifiable value of damping.

Both types of analyses verify integrity of the equipment is maintained under low-level earthquake loads including appropriate RBV dynamic loads in combination with normal operating loads and normal operating and DBE loads including appropriate RBV dynamic loads, unless otherwise justified.

See Section 3.3.1.3 for additional details and discussion of Seismic Analysis of Seismic Category A and B Subsystems.

#### **3.9.3.4 Seismic Qualification by Combined Testing and Analysis**

Qualification by combined testing and analysis is used as a method for qualification for complex or large equipment where it is not practical to test the entire assembly or it is too large to be tested at once, unless another method of qualification is justified.

One method of combined qualification is to use a representative prototype portion or scaled-down prototype of the assembly that is subjected to type testing. The data from the type testing is then used to develop and validate an analytical model of the prototype. The prototype analytical model is then extrapolated to represent the larger assembly and then using the results to justify qualification of the equipment based on prototype testing.

A second method of combined qualification is to mount the full assembly to a rigid floor to simulate service mounting and then a portable shaker test (or an impact or pull test if justified) is performed to excite the natural or resonance frequencies of the specimen. The amplification of resonance motion is used to determine the appropriate modal frequency and damping for a dynamic analysis of the equipment.

For equipment with multiple site configurations the combined qualification method can be applied to reduce the number of configurations to be tested. In this case, an evaluation must be performed to determine the enveloping “worst-case” configuration(s), which is then tested. Analysis is then used to justify the various configurations based on the “worst-case” configuration(s).

The combination method can be used for qualification of larger electrical equipment support assemblies containing Seismic Category A or B equipment. For this case, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range. If the support is determined to be free of natural frequencies in the critical frequency range, then it is assumed to be rigid and a static analysis is performed and calculations of transmissibility and responses to varying input accelerations are determined to see if Seismic Category A or B equipment mounted in the assembly would operate without malfunctioning.

#### **3.9.3.5 Beyond Design Basis Earthquake**

REGDOC-2.5.2 Section 7.13 (Reference 3.9-1) states that for a beyond-design-basis earthquake (BDBE), demonstration that there is a high confidence of low probability of failure (HCLPF) of the SSC that are credited to function during and after the event. This demonstration need not be seismic qualification by testing. BDBE is identified as a Checking Level Earthquake (CLE). Typically, the CLE (as discussed in Section 3.5.6.1.2) is considered a DEC. DEC for seismic events are a subset of beyond design basis seismic events that are considered in the evaluation of the facility using best-estimate methodology to keep releases of radioactive material within acceptable limits.

If determined to be useful, fragility testing per IEC/IEEE 60980-344 (Reference 3.9-7) may be used as a qualification method. Fragility testing is a form of vibration testing of an SSC to determine the point where it can no longer perform its function, whether due to electrical or mechanical malfunction, or excessive structural deformation or destruction. Where fragility testing is performed, it provides useful information about margin to failure. Knowledge of the seismic fragility of an SSC is useful in determining its seismic margin to failure and in providing determination of SSC functionality in BDBE evaluations (per CSA N289.1 (Reference 3.9-2).

Seismic PSA is used to analyze the plant response to seismic hazards as discussed in Chapter 15, Section 15.6.

#### **3.9.3.6 Documentation**

Seismic qualification documentation including identification of seismic equipment, test/analysis plans and reports, technical specifications, data sheets, engineering standards, and component specific seismic qualification parameters, and requirements for inspection, maintenance and procurement are prepared in an auditable summary report in accordance with Clause 7 of 289.4 (Reference 3.9-5).

Documentation of seismic testing is in accordance with CSA N289.4 Section 5.8 (Reference 3.9-5) and IEC/IEEE 60980-344, Section 13 (Reference 3.9-7) and include, at minimum, locations of accelerometers, any existing resonance frequency(s) and transmission ratios, equipment damping coefficients if there is resonance over the range of the test response spectra, test equipment used, any modifications made to test specimen, hardware interface requirements, test methods, approval signature and dates, description of test facility, summary of results, equipment seismic qualification conclusions (including RBV dynamic loads), anomalies and their resolution, test data, and justification for using single-axis or single frequency tests for all items that are tested in this manner.

### **3.9.4 Environmental Qualification**

#### **3.9.4.1 Scope**

Environmental Qualification is a subset of equipment qualification specifically addressing equipment exposure to a harsh environment. In alignment with CNSC REGDOC-2.5.2 Section 7.8 (Reference 3.9-1) and CSA N290.13 (Reference 3.9-9), Environmental Qualification is established to ensure that BWRX-300 SC1 SSC can perform their FSFs during and after exposure to a harsh environment resulting from a DBA during and after which they are required to operate. Equipment whose failure due to the harsh environment could impair the ability of qualified equipment to perform safety category functions are also considered for Environmental Qualification. Equipment that is not significantly impacted by the increased stress due to the harsh environment, or for which there are not credible failure modes induced by the harsh environment preventing the equipment from performing its FSF are exempt from Environmental Qualification. The effects of normal service conditions including that of AOOs, and the impact of aging are considered in the SSC ability to perform their safety category functions.

While Environmental Qualification is not required to be established for equipment responding to DECAs as stated in CSA N290.13 (Reference 3.9-9), equipment survivability assessments are used to provide reasonable confidence that equipment will function in response to the DEC within the time span required and that instrumentation will function with reasonable accuracy per REGDOC-2.5.2 (Reference 3.9-1). IEC/IEEE 60780-323 (Reference 3.9-8) provides considerations for qualifying equipment for DECAs and guidance is provided in Annex B of CSA N290.13 (Reference 3.9-9), and CSA N290.16 (Reference 3.9-10).

#### **3.9.4.2 Environment Parameters**

A harsh environment occurs as a result of a subset of DBAs for which ambient and operational service conditions change significantly as a result of the DBAs, DBAs considered in the BWRX-300 design are discussed in Chapter 15. Environmental parameters considered when screening for a harsh environment include:

- Temperature
- Steam
- Condensing Humidity
- Pressure
- Submergence
- Radiation
- Chemistry

Table 3.9-1 lists harsh environment screening criteria for environmental parameters based on the guidance in CSA N290.13 Annex A (Reference 3.9-10).

Per CSA N290.13, (Reference 3.9-10), a mild environment is one that would at no time be significantly more severe than the environment that would occur during the normal plant operation, including during AOOs, and would not give rise to significant aging mechanisms. For equipment located in a mild environment during and after a DBA for which it is required to function, Environmental Qualification is not required.

Per the description of mild environment qualification in CNSC REGDOC-2.5.2, Section 7.8 (Reference 3.9-1), for equipment not requiring Environmental Qualification per the scope of CSA N-290.13 (Reference 3.9-9) as described herein, the environmental conditions for its expected



function would be identified in its design specification and a manufacturers certification that the equipment meets the specification would be provided.

### **3.9.4.3 Objectives**

The objectives of Environmental Qualification of BWRX-300 SSC include:

1. Identification of SSC required to be environmentally qualified
2. Establishment of the safety category functions, performance requirements, normal service conditions, and post-accident harsh environment conditions for SSC identified as requiring qualification
3. Documentation of objective evidence verifying that the identified SSC are capable of performing credited safety category functions under the relevant harsh conditions, including consideration of age-related degradation during normal service
4. Controls and evidence to ensure that SSC are installed considering identified configuration and interface requirements
5. Controls and evidence to ensure that qualification of the equipment is preserved throughout the design life including aging and obsolescence

### **3.9.4.4 Requirements for Environmental Qualification**

#### **3.9.4.4.1 DBA Identification**

BWRX-300 DBAs that produce a harsh environment with potential to cause common cause failures are identified and analyzed at the appropriate design phase. Documentation of the basis for classifying an accident as harsh is included.

#### **3.9.4.4.2 Defining Normal and Accident Environmental Envelope**

At the appropriate design phase an environmental envelope that includes a listing of all areas of the facility in which SSC are expected to fulfill safety category functions during and after a DBA is identified and documented. For each identified area, the ambient environmental and operational conditions are provided for normal conditions (normal operating modes and AOOs), and for DBA conditions based on the limiting parameters identified from DBA identification.

#### **3.9.4.4.3 Identification of Equipment Requiring Harsh Environment Qualification**

At the appropriate design stage, BWRX-300 equipment requiring Environmental Qualification (as described in 3.9.4.1) is identified and documented. The list also includes equipment whose failure due to the harsh environment could impair the performance of qualified equipment. Equipment that is not significantly impacted by the increased stress due to the harsh environment, or for which there are not credible failure modes induced by the harsh environment preventing the equipment from performing its safety category function is exempt from Environmental Qualification. A basis for exempting equipment from qualification (e.g., failure modes, environmental conditions, materials, etc.) will be documented.

Information documented in the list of environmentally qualified equipment includes:

- Equipment identification
- Safety category function
- Applicable DBA
- Mission time
- Normal and accident service conditions

#### **3.9.4.4 Qualified Life**

Qualified life is established for equipment determined to be susceptible to age-related degradation for the specified service conditions. The equipment included within the scope of the Environmental Qualification program is analyzed based on an expected plant life of 60 years or is subject to replacement or evaluation of the effects of aging and obsolescence on a periodic basis.

#### **3.9.4.5 Establishing Environmental Qualification**

Methods for demonstration that equipment is environmentally qualified include testing, analysis, by operating experience, or by a combination of these methods in accordance with CNSC REGDOC-2.5.2 (Reference 3.9-1), CSA N290.13 (Reference 3.9-9), Reg. Guide 1.89 (Reference 3.9-11), and IEC/IEEE 60780-323, (Reference 3.9-8).

##### **3.9.4.5.1 Qualification By Testing**

Type testing is the preferred method for demonstrating that equipment is Environmentally Qualified. A type test subjects a representative sample of equipment, including interfaces, to a series of tests, and include simulating the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to conditions that simulate DBA harsh conditions and thereby establishes the tested configuration for installed equipment service, including mounting, orientation, interfaces, conduit sealing, and expected environments. A type test demonstrates that the equipment performs the intended safety category function(s) for the required operating time before, during, and/or following the DBA, as appropriate.

Type tests are performed in accordance with applicable industry standards, such as CSA N290.13 (Reference 3.9-9) and IEC/IEEE 60780-323 (Reference 3.9-8).

A typical sequence includes, but is not limited to the following:

- Initial inspection
- Baselines functional test
- Normal radiation exposure
- Accident radiation exposure
- Accelerated thermal aging
- Other aging simulation as applicable
- Post-aging functional test
- Accident simulation
- Final inspection

##### **3.9.4.5.2 Qualification by Analysis**

Qualification by analysis requires the construction of a valid mathematical model of the equipment to be qualified, in which the performance characteristics of the equipment are dependent variables, and the environmental influences are the independent variables. The validity of the mathematical model is justified by test data, operating experience, vendor data, and established engineering principles that support the analytical assumptions and conclusions.

Consistent with CSA N290.13 (Reference 3.9-9), the qualification of complex equipment by analysis only is not used because of the great difficulty in developing an accurate analytical model, unless it can be justified that using only analysis is sufficient.

#### **3.9.4.5.3 Qualification by Operating Experience**

Qualification by use of operating experience requires documented data to be available confirming that the product providing the operating experience is identical or justifiably similar to the equipment to be qualified, the product providing the operating experience has operated under service conditions which equal or exceed, in severity, the service conditions and performance requirements for which the product is to be qualified, and the installed product must, in general, be removed from service and subjected to partial type testing to include the DBA environments for which the product is to be qualified. Operating experience may also provide information on limits of extrapolation, failure modes, and failure rates.

#### **3.9.4.5.4 Combined Qualification**

Equipment may be qualified by test, analysis, operating experience, or any combination of these methods. Combined qualification may be used to supplement existing test data. Partial type testing may be augmented by tests of components where size, applications, time, or other test limitations preclude the use of a full type test. Examples of combined qualification include separate effect tests with extrapolation or analysis, operating experience with extrapolation or analysis, and type tests supplemented with tests of components and analysis.

#### **3.9.4.5.5 Aging Considerations**

Significant aging mechanisms are considered in establishing Environmental Qualification for the specified service conditions and in defining the qualified life of equipment and components. An aging mechanism is significant if subsequent to manufacture, while in storage, and/or in the normal and abnormal service environment, it results in degradation of the equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety category function under harsh environmental DBA conditions. These typically include thermal, radiation, and operation induced degradation. Age conditioning is used during qualification to simulate these effects.

Accelerated thermal aging is used to simulate the deterioration due to temperature during the normal service life of equipment. The use of the Arrhenius Equation is the recognized method.

The effects of radiation are simulated during qualification testing for equipment exposed to radiation in normal or accident conditions. Radiation qualification considers that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification includes doses from all potential radiation sources at the equipment location. The assessment of accelerated aging effects due to normal radiation exposure is performed separately from or included as part of the accident radiation exposure.

Cycle aging conservatively simulates the degradation during the required operating cycles for the equipment. The number of cycles required for equipment is based on the design specification.

For equipment that cannot meet the required cycles for the 60-year life, a shorter qualified life is established, and the effects of physical aging and obsolescence are reflected in the maintenance, surveillance, and replacement program.

Age conditioning considers sequential, simultaneous, and synergistic effects to achieve the worst state of degradation.

Age conditioning is not required for equipment with no determined aging mechanisms.

#### **3.9.4.5.6 Environmental Margins**

Margin is applied during Environmental Qualification to account for unquantified uncertainties such as normal variations in equipment production, inaccuracies in measurement and test instrumentation and reasonable errors in defining satisfactory performance. Current qualification practices do not require statistical or reliability data when establishing Environmental Qualification. Instead, conservatism and margins are intended to provide reasonable assurance that the installed equipment can perform as required.

The following margins as recommended in CSAN290.13 (Reference 3.9-9) may be applied to simulated accident conditions during qualification testing or considered when performing qualification by analysis.

The margin applicable to a specific parameter is determined based on the peak conditions as follows:

- Temperature: + 10% of peak temperature to a maximum of 8°C
- Pressure: + 10% of peak gauge pressure to a maximum of 70kPa
- Radiation: + 10% of the total integrated accident dose
- Mission Time: + 10% of the required mission time (up to the maximum)

#### **3.9.4.6 Documentation of Environmental Qualification**

Documentation is required to ensure an auditable proof of performance under DBA conditions is developed and maintained for equipment requiring Environmental Qualification. The following subsections provide a general description of the expected information. The organization or format of the documentation is not intended to be prescriptive.

##### **3.9.4.6.1 Equipment Specifications for Environmental Qualification**

Plant specific equipment specifications for Environmental Qualification are developed and include essential information about the equipment to be qualified. The following is included as applicable:

- Details of aging stressors resulting from normal environmental conditions
- Details of aging stressors resulting from normal operating conditions
- Details of in-plant configuration, including mounting
- Description of control, indication, and other auxiliary devices required for proper operation
- Functional requirements under the defined normal and accident service conditions
- Required qualified life for the equipment or maintenance intervals for specific components, or both
- Details of DBA stressors resulting from accident environmental conditions
- Details of DBA stressors resulting from accident operating conditions
- Performance requirements and acceptance criteria
- Mission time(s) for relevant safety category functions of equipment
- Provision for condition monitoring

#### **3.9.4.6.2 Qualification Plan**

Prior to starting the qualification of equipment, plans are developed detailing the qualification method. If the qualification method is by test, the qualification plan is incorporated into the test plan. The following is included:

- Equipment identification
- Equipment qualification specifications requirements for Environmental Qualification as described above
- Scope of qualification
- Documentation for traceability of equipment and of all polymeric or elastomeric material
- A description of the components of the equipment
- Qualification method selected and justification for the selection of a method if it is other than testing
- When analysis is the chosen method, a description of the analytical methods to be used
- Age conditioning limits/parameters, including qualified life objective, peak aging temperature limits, radiation dose, and condition-based qualification methods, if applicable
- Evaluation of identified synergistic effects

#### **3.9.4.6.3 Test Report**

For qualification by test, a test report is developed after the completion of testing.

A test report includes the test plan and provides a detailed summary of the testing performed and the test results to demonstrate the equipment is successfully qualified for the environmental conditions specific to the testing. As a minimum it includes:

- Approved and dated certification sheet
- Identification of equipment tested
- Identification of test specimen
- The range of types or sizes covered
- The qualification requirements
- Results of initial and final inspection
- Description of mounting configuration during testing
- The simulated aging and accident environmental conditions as a function of time
- Results of all functional tests
- A description of the test facility
- A description of the test facility's QA program
- Calibration details for test equipment
- Disposition of any anomalous test results and variance from the test plan
- Details of any maintenance performed

- A summary of the testing program
- A conclusion stating compliance/non-compliance with acceptance criteria and test plan
- Details of connections and interfaces with the tested equipment
- A determination of the qualified life of the equipment under specified service conditions

#### **3.9.4.6.4 Analysis Report**

For qualification by analysis, an analysis report is developed providing a detailed summary of the analytical method used (including identification of any software used), calculations performed, and the results to demonstrate the equipment is successfully qualified for the environmental and/or seismic/dynamic condition(s) specified by the analysis.

#### **3.9.4.6.5 Qualification Summary Report**

An Environmental Qualification summary report provides documented assurance in an auditable format that equipment requiring Environmental Qualification should function as required under the relevant service conditions for its required mission time. It establishes the basis for equipment configuration, maintenance and procurement requirements providing the means to ensure that Environmental Qualification of the equipment is maintained for the station's life. Information contained in the summary report includes:

1. Equipment identification and description including function, location, mounting and interfaces, any required enclosures/shielding consistent with qualification basis
2. The qualification basis for the equipment including methodology, documentation from testing, analysis, and other supporting documentation supporting qualification
3. An overall conclusion on the qualified status of the equipment, including any limitations on use, operating constraints, or restrictions
4. Identification of any specific maintenance, replacement, and surveillance activities necessary to ensure that the qualification of the equipment is preserved throughout its installed life
5. Identification of any specific procurement requirements necessary to ensure that replacement equipment or components are procured in a manner that is consistent with the qualification basis
6. Identification of any handling and storage requirements

### **3.9.5 Electromagnetic Compatibility**

Accepted industry codes and standards are applied to establish an electromagnetic compatible environment applicable to electrical and I&C equipment. EMC qualification involves two elements:

1. Testing to assess susceptibility of equipment to interference levels that bound the expected electromagnetic environment
2. Testing to assess emissions of equipment to ensure that the contribution to the electromagnetic environment does not invalidate bounding interference levels applied for susceptibility testing

Susceptibility testing allows assessment of equipment immunity to Electromagnetic and Radio-Frequency Interference (EMI/RFI) and confirmation of its Surge Withstand Capability. Emissions testing provide assurance that equipment is compatible with the expected electromagnetic environment.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Consistent with CNSC REGDOC-2.5.2 (Reference 3.9-1), EMI/RFI is addressed through recognized industry standards. NRC Reg Guide 1.180 (Reference 3.9-12) provides appropriate guidance for the EMC testing, describing methods and procedures considered acceptable for demonstrating EMC compliance based on the endorsement of IEC standards IEC 61000-2 / 4 (References 3.9-13 and 3.9-14), Military Standards MIL-STD (Reference 3.9-15) EPRI Topical Report TR-102323 (Reference 3.9-16) and IEEE Standard 627 (Reference 3.9-17) for test methods consistent with specific equipment requirements.

Chapter 2, Section 2.2.9 characterizes the site-specific electromagnetic hazards for which the design must consider and for which EMC qualification must address.

Chapters 7 and 8 describe the design of the I&C systems and the Electrical systems, respectively. As part of the design process, layout strategies are developed to ensure that the design considers interaction between SSC, and as the design is constructed, elements such as grounding and shielding are incorporated to meet the EMC/EMI standards (prior to testing).

Chapters 7, Subsections 7.3.1.3.1, 7.3.2.3.1, and 7.3.4.3.1 discuss design and quality measures for I&C systems as they relate to qualification measures that confirm I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the plant state and for the area in which they are located. Chapter 7, Table 7.1-1 provides System and Equipment standards to be followed in the design that ensures qualification measures are applied.

Chapter 8, Section 8.6 provides electrical system design information on grounding and EMC. Chapter 8, Section 8.1.1.2 describes how electrical systems are designed to accommodate grid disturbances. The electrical design includes considerations for the environmental conditions postulated for plant states in the areas in which components are located and credited to function.

The standards referenced provide detailed test conditions to ensure equipment is tested in the environments in which they are expected to function and provide post-installation practices for maintaining qualification including handling and storage requirements.

### **3.9.6 Specific Equipment Requirements**

Specific equipment categories may have additional requirements not applicable generically across all qualification programs. The Electrical and I&C equipment must meet the guidance provided in CSA N289 series (References 3.9-2 through 3.9-6) and the CSA N290 series standards (Reference 3.9.18 through Reference 3.9-22).

#### **3.9.6.1 Mechanical Equipment**

Safety Class mechanical equipment, which has the sole safety category function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME Boiler and Pressure Vessel Code, Section III (Reference 3.9-23), is considered qualified as specified in CSA N290.13 (Reference 3.9-9).

Mechanical equipment can be qualified by presenting historical performance data if it is demonstrated that the equipment satisfactorily sustains dynamic loads which are equal to or greater than those specified for the equipment and that the equipment performs a function equal to or better than that for which it is specified.

For mechanical equipment where the loading under normal service is more severe than loading under DBA, then the loading under normal service must be considered in addition to the loading under DBA by test and/or analysis.

For mechanical equipment, the loading and capability under DBA conditions is analyzed in the qualification process to establish the suitability of materials, parts, and equipment needed for

safety category functions, and to verify that the design of such materials, parts, and equipment is adequate.

The qualification of mechanical equipment includes, as applicable, materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment, the environmental conditions and process parameters for which this equipment must be qualified, non-metallic material capabilities, and the evaluation of environmental effects.

In addition, the qualification guidance provided in ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, (Reference 3.9-24), is considered for qualification of active mechanical equipment. Mechanical pipe supports of SC1 equipment that are susceptible to environmental degradation are seismically and environmentally qualified.

### **3.9.6.2 Electrical Equipment**

Additional qualification guidance is considered for specific electrical equipment, if applicable, as follows:

- SC1 Batteries and their supporting element – IEEE 535 (Reference 3.9-25)
- SC1 Transformers IEEE 638 – (Reference 3.9-26)
- Static battery chargers and inverters – IEEE 650 (Reference 3.9-27)
- Electric penetration assemblies – IEEE 317 (Reference 3.9-28)
- SC1 Actuators – IEEE 382 (Reference 3.9-29)
- SC1 Continuous duty motors – IEEE 334 (Reference 3.9-30), as endorsed by Reg Guide 1.40 (Reference 3.9-31)
- SC1 Motor Control Centers (MCCs) – IEEE 649 (Reference 3.9-32)
- For the electrical equipment described above, excluding motors, the EMC qualification guidance provided in Reg Guide 1.180, (Reference 3.9-17) is considered

### **3.9.6.3 Instrumentation & Control Equipment**

Additional qualification guidance is considered for specific I&C equipment, if applicable. For example, control boards, panels, and racks classified as SC1 components utilize IEEE 420, (Reference 3.9-33) for their qualification program.

Qualification of computer-based I&C systems is in accordance with CNSC REGDOC-2.5.2, (Reference 3.9-1), CSA N290.13 (Reference 3.9-9), and IEEE 7-4.3.2 (Reference 3.9-34) which is consistent with the EMC requirements specified in Reg Guide 1.180 (Reference 3.9-12) and described in Subsection 3.9.5.

When computer based I&C systems environmental type testing is performed:

1. The system under test demonstrates that it functions and performs with safety software that has been validated and verified and is representative of the software to be installed in-service.
2. The testing demonstrates performance of all safety category functions that may be impacted by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, that are tested during verification and validation testing, are not required to be tested unless their outputs exercise different hardware components which may be impacted by environmental conditions.



3. The testing exercises all portions of the system that are necessary to accomplish the safety category functions and those portions whose operation or failure could impair the safety category functions.
4. The testing confirms the response of digital interfaces and verify that the design accommodates the potential impact of environmental effects on the overall response of the system.

The testing of a complete system is preferred. When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety performance.

#### **3.9.6.4 Cables, Raceways, Supports, etc.**

For qualification of SC1 cables, the qualification guidance provided in CSA N290.13, (Reference 3.9-10) and IEEE 383 (Reference 3.9-35) are considered.

Supports (hangers) that support trays or conduit that carry safety circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628 (Reference 3.9-36).

Supports used for Non-Safety Class raceway (conduit and cable tray) in Seismic Category A structures are analyzed to withstand the effects of a DBE and evaluated for seismic interaction as applicable.

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, (Reference 3.9-37) as endorsed by Reg Guide 1.156, (Reference 3.9-38) for their qualification program.

#### **3.9.6.5 Line-Mounted Equipment**

Guidance in IEEE 572 (Reference 3.9-37) and IEC/IEEE 60980-344 (Reference 3.9-9.) identifies that special consideration is required for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition will occur as a result of the piping or duct system.

Guidance and further clarification for special considerations for line-mounted equipment is provided in IEEE 572 (Reference 3.9-33) and IEC/IEEE 60980-344 (Reference 3.9-8) as well as IEEE 382 (Reference 3.9.10.11-29).

#### **3.9.7 References**

- 3.9-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.9-2 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.9-3 CSA N289.2, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.9-4 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.9-5 CSA N289.4, "Testing procedures for seismic qualification of nuclear power plant structures, systems, and components." CSA Group.
- 3.9-6 CSA N289.5, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities," CSA Group.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.9-7 IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.9-8 IEC/IEEE 60780-323, "Nuclear facilities – Electrical equipment important to safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.9-9 CSA N290.13, "Environmental qualification of equipment for nuclear power plants," CSA Group.
- 3.9-10 CSA N290.16, "Requirements for beyond design basis accidents," CSA Group.
- 3.9-11 USNRC Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."
- 3.9-12 USNRC Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems".
- 3.9-13 IEC 61000-6-2, "Electromagnetic compatibility (EMC) – Part 6-2: Generic standards – Immunity standard for industrial environments," International Electrotechnical Commission.
- 3.9-14 IEC 61000-4, "Electromagnetic Compatibility (EMC) – Part 4: Testing," International Electrotechnical Commission.
- 3.9-15 MIL-STD-461G, "Electromagnetic Interference Characteristics of Equipment," US Department of Defense.
- 3.9-16 EPRI TR-102323, "Guidelines for Electromagnetic Interference Testing in Power Plants," Electric Power Research Institute.
- 3.9-17 IEEE 627, "Standard for Qualification of Equipment Used in Nuclear Facilities", Institute of Electrical and Electronic Engineers.
- 3.9-18 CSA N290.0, "General requirements for safety systems of nuclear power plants," CSA Group.
- 3.9-19 CSA N290.14, "Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants," CSA Group.
- 3.9-20 CSA N290.4, "Requirements for reactor control systems of nuclear power plants," CSA Group.
- 3.9-21 CSA N290.7, "Cyber Security for Nuclear Facilities," CSA Group.
- 3.9-22 CSA N290.8, "Technical specification requirements for nuclear power plant components," CSA Group.
- 3.9-23 ASME BPVC-III, "Boiler and Pressure Vessel Code Section III - Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.9-24 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.
- 3.9-25 IEEE 535, "Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-26 IEEE 638, "Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.9-27 IEEE 650, "Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-28 IEEE 317, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-29 IEEE 382, "Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants," Institute of Electrical and Electronic Engineers.
- 3.9-30 IEEE 334, "Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-31 USNRC Regulatory Guide 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants."
- 3.9-32 IEEE 649, "Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-33 IEEE 420, "Standard for the Design and Qualification of Class 1E Control Boards, Panels and Racks Used in Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-34 IEEE 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-35 IEEE 383, "Standard for Qualifying Class 1E Electric Cable and Field Splices for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-36 IEEE 628, "Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-37 IEEE 572, "Standard Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-38 USNRC Regulatory Guide 1.156, "Qualification of Connection Assemblies for Nuclear Power Plants."

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 3.9-1: Harsh Environment Parameter Conditions**

Parameter		Condition
Temperature		10°C above normal ambient and $\geq 50^{\circ}\text{C}$ <sup>(1)</sup>
Pressure		>4 kPa(g) (or 10%) increase or decrease from normal ambient pressure due to a DBA <sup>(2)</sup>
Humidity		100% Relative Humidity or condensing steam conditions <sup>(3)</sup>
Submergence		Any <sup>(4)</sup>
Radiation	Non-electronic equipment	DBA Total integrated accident dose (TIAD) > 170 Gy (17 krad) <sup>(5)</sup>
	Electronic equipment	TIAD > 10 Gy (1krad) <sup>(6)</sup>
Chemistry		Significant change in chemistry of the ambient environment or operating conditions

- (1) Temperature criteria are based on 10°C as a significant increase in normal ambient temperature added to the typical 40°C ambient temperature rating of most industrial EI&C equipment.
- (2) Typically, pressure change must be coincident with other DBA stressors to be considered harsh.
- (3) If steam is present under normal conditions, it is not a harsh DBA stressor. If condensing humidity condition do not change following a DBA, it is not a harsh DBA stressor.
- (4) Submergence is not harsh if it also occurs under normal operation.
- (5) Based on the radiation threshold of the most radiation-sensitive polymer.
- (6) Based on the radiation threshold of integrated circuits.

### **3.10 In-Service Monitoring, Tests, Maintenance, and Inspections**

#### **3.10.1 Safety Design Bases and Requirements**

Ontario Power Generation DNNP-1 Project Quality Plan identifies the controls and describe the quality requirements to be implemented throughout the development of the BWRX 300 SMR project. This Project Quality Plan supplements NEDO 11209-A (Reference 3.10-12), for the execution of GEH design activities that are associated with the BWRX-300 project. NEDO 11209-A has been approved by the U.S. Nuclear Regulatory Commission (NRC). In addition, the CSA Group (CSA) Standard N299 Series (Reference 3.10-7 Thru 3.10-9) defines a consistent set of Canadian quality assurance program requirements for the provision of items and services for nuclear power plants.

The Canadian Nuclear Safety Commission (CNSC) governs the Canadian nuclear industry regulations and has jurisdictional authority. Canadian suppliers comply with CNSC regulations. U.S. based suppliers who export to Canada may request a waiver from U.S. CFRs, RGs, and NUREG and comply with CNSC regulations. In addition, CSA Standards N299.1, N299.2, and N299.3, defines the Canadian quality assurance program requirements for the provision of items and services for nuclear power plants, Categories 1, 2, and 3, respectively.

CNSC REDOC 2.6.1 (Reference 3.10-17), Section 3, is used as guidance for establishment of inspections, tests, modeling, and monitoring programs for the DNNP BWRX-300 Nuclear Power plant. Chapter 13 provides the specific features of the programs.

CNSC REGDOC-2.5.2, Version 1 (Reference 3.10-16) and CNSC REGDOC-2.6.2 (Reference 3.10-18) provide the primary requirements for addressing In-Service Monitoring, Tests, Maintenance, and Inspections.

SSCs that have shorter service lifetimes than the plant lifetime will be identified and described in the design documentation.

Design requirements associated with In-service Monitoring, Tests, Maintenance, and Inspections involve accessibility, ALARA, aging management and easy-removable insulation for inspection, testing, and maintenance. In cases where SSCs are of safety class and cannot be designed to support the desirable testing, inspection, or monitoring schedules, one of the following approaches shall be taken:

1. Proven alternative methods, such as surveillance of reference items or use of verified and validated calculation methods, shall be specified.
2. Conservative safety margins shall be applied, or other appropriate precautions shall be taken, to compensate for possible unanticipated failures.

#### **3.10.2 In-Service Monitoring**

The BWRX-300 levels of in-service monitoring for SSC is related to the Defence-in-Depth Defense Levels (DL) that are specified in Section 3.1 and associated classifications of SSCs in Section 3.2. Specifics on In-service monitoring are developed in the other PSAR chapters.

The design provides facilities for monitoring chemical conditions of fluids and of metallic and non-metallic materials. The means for adding or modifying the chemical constituents of fluid streams is specified in Chapter 13, Subsection 13.3.2.3 programmatic requirements for in-service monitoring.

#### **3.10.3 In-Service Testing**

IST of certain ASME Boiler and Pressure Vessel Code (BPVC) Section III Division 1 (Reference 3.10-1) pumps, valves, and snubbers (dynamic restraints) as applicable is performed in

accordance with the ASME OM code. In addition, IST is performed in accordance with applicable Canadian Codes and Standards, and IAEA Safety Standards.

Pre-service test results will be documented and used as a baseline for periodic in-service testing.

The design of BWRX-300 structures, systems, and components provides access for the performance of IST to the extent practicable.

The IST Program includes periodic tests and inspections that demonstrate the operational readiness of certain SSC that perform a function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident.

Specific required in-service tests are established in other PSAR chapters involving SSCs.

Chapter 13, Subsection 13.3.2.3, provides programmatic requirements for in-service testing.

#### **3.10.4 In-Service Maintenance**

CNSC REGDOC-2.6.2 (Reference 3.10-16) forms the regulatory bases for the requirements of the Canadian Nuclear Safety Commission (CNSC) regarding maintenance programs for nuclear power plants (NPPs). This document also provides information and guidance on how the requirements may be met. The DNNP BWRX-300 Nuclear Power plant will abide by the recommendations of CNSC REGDOC-2.6.2 which are based in part on the following publications:

- CNSC, REGDOC-2.6.1, Reliability Programs for Nuclear Power Plants (Reference 3.10-15).
- CNSC, REGDOC-2.5.2, Version 1, Design of Reactor Facilities: Nuclear Power Plants (Reference 3.10-14).
- CNSC, REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Reactor Facility, Version 2 (Draft) (Reference 3.10-13).
- International Atomic Energy Agency (IAEA), TECDOC-658, Safety Related Maintenance in the Framework of the Reliability Centered Maintenance Concept, Vienna, 1992 (Reference 3.10-10).
- IAEA Safety Standards Series, No. NS-G-2.6, Maintenance, Surveillance, and In-service (Reference 3.10-11).
- CSA N286-12, Management system requirements for nuclear facilities (Reference 3.10-6).

Baseline data will be gathered during initial testing and system commissioning of SSCs.

Chapter 13, Subsection 13.3.3, provides programmatic requirements for in-service maintenance.

#### **3.10.5 In-Service Inspection**

Mechanical components and equipment including heat exchangers, pipe supports, pumps, valves, and vessels, that are classified as ASME BPVC III Division 1 Class 1, 2, or 3 are designed and provided with accessible openings for ISI and testing, to justify the operational readiness of components and equipment as set forth within ASME BPVC III- Division 1.

Components and equipment, that require inspections and testing to satisfy ASME BPVC-XI-Division 1 requirements, are examined by appropriate ISI and testing techniques, including ASME BPVC III Division 1, ASME Code OM, CNSC REGDOC-2.5.2, and CNSC REGDOC 2.6.2 required examinations, prior to the component or equipment leaving the manufacturer's facility.

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

ASME BPVC-XI-2021, ASME Code OM, CNSC REGDOC-2.5.2, and CNSC REGDOC 2.6.2 inspection and testing requirements do not replace or change ASME BPVC III required examinations.

Nondestructive Examination (NDE) methods are described within ASME BPVC-V (Reference 3.10-2) and ASME BPVC-XI.

Component and equipment procurement specifications provide detailed requirements, which are to be used during the manufacturing phase and installation at the plant site.

Chapter 13, Subsection 13.3.2.3, provides programmatic requirements for ISI.

### **3.10.6 References**

- 3.10-1 ASME BPVC-III, "Boiler and Pressure Vessel Code Section III - Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.10-2 ASME BPVC-V, "Section V - Non-destructive Examination," American Society of Mechanical Engineers.
- 3.10-3 ASME BPVC-XI, "Boiler and Pressure Vessel Code Section XI - Rules for In-Service Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.10-4 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 3.10-5 ASME OM, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers.
- 3.10-6 CSA N286-12, "Management System Requirements for Nuclear Facilities," CSA Group.
- 3.10-7 CSA N299.1-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 1," CSA Group.
- 3.10-8 CSA N299.2-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 2," CSA Group.
- 3.10-9 CSA N299.3-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 3," CSA Group.
- 3.10-10 IAEA TECDOC-658, "Safety Related Maintenance in the Framework of the Reliability Centered Maintenance Concept," International Atomic Energy Agency.
- 3.10-11 IAEA Safety Standards Series No. NS-G-2.6, "Maintenance, Surveillance, and In-service Inspection in Nuclear Power Plants," International Atomic Energy Agency.
- 3.10-12 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.10-13 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.10-14 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.10-15 CNSC Regulatory Document REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants."
- 3.10-16 CNSC Regulatory Document REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants."

### **3.11 Compliance with National and International Standards**

Chapter 1, Appendix B Tables B1-11 through B1.11-3 Conformance with Applicable Regulations, codes, and standards, describes the applicable CNSC Regulatory documents, codes and standards used in the design of the OPG DNNP BWRX-300 plant. CNSC REGDOC 1.1.2 Draft Version 2 and CNSC REGDOC 2.5.2 Draft Version 2 form the basis of the Canadian regulatory requirements. The CSA Group (CSA) standards form the detailed bases of code and standard methodology to comply with the regulatory requirements and compared to the standards (both National and International) used in the BWRX-300 design. Many CSA standards refer to the use of U.S. codes in the design of Canadian Nuclear Plants. Alternative codes, standards, and methodology not addressed by CSA standards are reviewed against CNSC REGDOC requirements and justified through a design assessment process for use. Chapter 17 on Safety in Design discusses the overall design process.

As stated in Chapter 1, section 1.11, CNSC Regulatory Documents, applicable IAEA and U.S. regulatory documents, and industry codes and standards used in the OPG BWRX-300 design, grouped by Safety and Control Area (SCA), are listed in Appendix B Tables B1.11-1 through 1.11-3. These tables represent all 14 SCAs that form the bases for CNSC safety reviews. The tables list the codes and standards by the organization that represents the applicability to design type such as Mechanical, Electrical, Civil, Nuclear I&C and others. The tables clarify any specific details associated with the code and/or standard use.

The specific PSAR chapters provide prescriptive details that related to the BWRX-300 design features and their alignment with Canadian regulations including compliance with both national and international standards. Chapter 3, Safety Objectives and Design Rules for Structures, Systems and Components forms the majority of requirements for other chapters used in the design of the DNNP BWRX-300 new nuclear plant.

#### **3.11.1 References**

- 3.11-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.11-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."



## **APPENDIX 3A – PRELIMINARY CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS**

### **3.12 Introduction**

The BWRX-300 approach to classifying Structures Systems and Components (SSC) is consistent with IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design" (Reference 3-12-1) and IAEA SSG-30, Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," (Reference 3.12-2) and aligns with CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," Section 7.1 (Reference 3.12-3). Classification of SSC is conducted to identify the importance of the SSC with respect to safety.

The methodology for classification of BWRX-300 SSC is discussed in Section 3.2. in accordance with:

- Safety Class (SC)
- Seismic Category
- Quality Group

Table 3.12-1 provides a preliminary list of the principal BWRX-300 components organized by system. Classification of Structures is presented in Section 3.3, Table 3.3-1.

#### **3.12.1 References**

- 3.12-1 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.12-2 IAEA Safety Standards Series No. SSG-30, "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," International Atomic Energy Agency.
- 3.12-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.12-4 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.12-5 ISO 9001, "Quality Management Systems - Requirements," International Organization for Standardization."
- 3.12-6 CSA N286-12, "Management System Requirements for Nuclear Facilities," CSA Group.
- 3.12-7 USNRC Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."
- 3.12-8 USNRC NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue of Reactor Materials."
- 3.12-9 10 CFR 21, "Reporting of Defects and Noncompliance."
- 3.12-10 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- 3.12-11 10 CFR 20.1201, "Occupational dose limits for Adults."

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

- 3.12-12 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 3.12-13 CSA N288.2, "Guidelines for Calculating the Radiological Consequences to the Public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents," CSA Group.
- 3.12-14 CSA N288.1:14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities," CSA Group.
- 3.12-15 USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning."
- 3.12-16 USNRC IN96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly."
- 3.12-17 ANSI/ANS-5.1, "American National Standard Decay Heat Power in Light Water Reactors," American Nuclear Society.
- 3.12-18 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 3.12-19 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components - Appendices," American Society of Mechanical Engineers.
- 3.12-20 ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13," American Society of Mechanical Engineers.

**Table 3.12-1: Preliminary BWRX-300 Classification List**

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
<b>NUCLEAR STEAM SUPPLY SYSTEMS</b>				
<b>Nuclear Boiler System</b>				
Reactor pressure vessel	SC1	SCCV	A	A
Main Steam (MS), Head Vent, Isolation Condenser System (ICS), Feed Water (FW), and Reactor Water Cleanup System (CUW) Reactor Isolation Valves (RIV)	SC1	SCCV	A	B
Core Support Structures: <ul style="list-style-type: none"> <li>• Shroud</li> <li>• Chimney</li> <li>• Core Support Ring and Legs (Shroud Support)</li> <li>• Core Plate (and Core Plate Hardware)</li> <li>• Top Guide (and Top Guide Hardware)</li> <li>• Orifice Fuel Supports and Peripheral Fuel Supports</li> <li>• Control Rod Guide Tubes (CRGTS)</li> <li>• Non-Pressure Boundary Portion of Control Rod Drive Housings (CRDHs)</li> </ul>	SC1	SCCV	B	A
Internal Structures: <ul style="list-style-type: none"> <li>• Nuclear Instrumentation In-Core Guide Tubes</li> <li>• Non-Pressure Boundary Portion of In-Core Housings</li> </ul>	SC1	SCCV	B	A
Internal Structures: <ul style="list-style-type: none"> <li>• Chimney Head and Steam Separator Assembly</li> <li>• Steam Dryer Assembly</li> <li>• Feedwater Spargers</li> <li>• Head Vent Internal Piping</li> <li>• CUW Suction Piping</li> <li>• Nuclear Instrumentation In-Core Guide Tube Stabilizers</li> <li>• ICS Return Internal Piping</li> </ul>	SC3	SCCV	B	NS
Surveillance Assembly (Sample Holders)	SCN	SCCV	NA	NS
Nuclear Instrumentation Dry Tube	SC1	SCCV	A	A
Nuclear Instrumentation Housings, Flanges and Ceramic Plugs	SC1	SCCV	A	A

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Pressure Boundary Portion of Control Rod Drive Housings	SC1	SCCV	A	A
Control Rods	SC1	SCCV	NA	B
Reactor Pressure Vessel (RPV) Support - Refueling Bellows	TBD	SCCV	TBD	TBD
RPV Stabilizers	SC1	SCCV	A	A
RPV Support Skirt	SC1	SCCV	A	A
Main Steam piping from the Reactor Isolation Valve to the outboard MS Containment Isolation Valve	SC1	SCCV	B	A
Outboard MS Containment Isolation Valves	SC1	RB	B	B
RPV Level Instrumentation Sensing Line including pressure retaining parts of instrumentation located on these lines	SC1	RB	B	A
MS line piping and components from outside the CIV to the Seismic Interface Restraint	SC1	RB	B	A
MS Seismic Interface Restraint	SC1	RB	B	A
MS line piping and components from the Seismic Interface Restraint (SIR) to the Condensate and Feedwater System, Main Turbine Equipment, Moisture Separator Reheater System, Turbine Bypass System, and Main Condenser and Auxiliaries components	SC3	TB	D	NS
MS line leak detection instrumentation in Reactor Building	SC1	RB	NA	B
MS line leak detection instrumentation in Turbine Building	SC1	TB	NA	NS
RPV Head Vent piping to MSL	SC1	SCCV	B	A
RPV Head Vent piping to Quench Tank Isolation Valve	SC1	SCCV	B	A
Quench Tank Isolation Valves	SC1	SCCV	B	B
RPV Head Vent piping from Quench Tank Isolation Valve to Quench Tank	SC3	SCCV	D	NS
Quench Tank	SC3	SCCV	D	NS
Head Vent Quench Tank Vacuum Breaker	SC3	SCCV	D	NS
O-Ring Seal Leak Detection piping up to Pressure Transmitter	SC3	SCCV	B	A See Note 8
O-Ring Seal Leak Detection piping to O-Ring Seal Leak Detection Manual Isolation Valve	SC3	SCCV	B	A See Note 8

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
O-Ring Seal Leak Detection Isolation Valves	SC3	SCCV	B	B See Note 8
O-Ring Seal Leak Detection Isolation Valve piping to Quench Tank	SC3	SCCV	D	NS
Other Nuclear Boiler System (NBS) mechanical / instrumentation ASME Section III pressure boundary components on the MS Lines	SC1	RB	B	A
Other NBS mechanical / instrumentation ASME B31.1 pressure boundary components on the MS Lines	SC3	TB	D	NS
<b>INSTRUMENTATION AND CONTROL SYSTEM</b>				
SC1 Instrumentation and Control System	SC1	RB and CB	NA	B
<i>SC2 and 3 Instrumentation and Control System</i>				
Equipment that supports DL2 functions	SC3	RB, TB, and CB	NA	NS
Equipment that supports DL4a functions	SC2	RB, TB, and CB	NA	NS
Equipment that supports DL4b functions	SC3	TBD	NA	NS
Non-Safety Instrumentation and Control System	SCN	RB, TB, and CB	NA	NS
<b>RADIATION MONITORING SYSTEMS</b>				
<b>Process Radiation and Environmental Monitoring System</b>				
<i>Process Radiation and Environmental Monitoring System, Process Radiation Monitoring Subsystem</i>				
In-line (external) radiation monitoring equipment (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	NA	NS
Off-line (process stream) radiation monitoring equipment (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	D	NS
<i>Process Radiation and Environmental Monitoring System, Area Radiation Monitoring Subsystem</i>				
Refueling Floor radiation monitors supporting Defense Line 2 functions (supporting PAM Type E variables)	SC3	RB	NA	NS
General Area radiation monitors (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	NA	NS
<i>Process Radiation and Environmental Monitoring System, Containment Monitoring Subsystem</i>				
CIVs and inboard process piping	SC1	RB	B	B

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment hydrogen and oxygen monitoring equipment (including process piping outboard of CIVs) (supporting PAM Type C and F variables)	SC3	RB	D	B
Containment fission product monitoring equipment (including process piping outboard of CIVs)	SC3	RB	D	NS
Containment water level transmitters	SC3	RB	NA	NS
Containment pressure transmitters supporting Defense Line 3 functions (supporting PAM Type C and D variables)	SC1	RB	NA	B
Containment pressure transmitters supporting Defense Line 4a functions	SC2	RB	NA	NS
Containment temperature transmitters (supporting PAM Type D variables)	SC3	RB	NA	B
Containment area radiation monitors (supporting PAM Type C and E variables)	SC3	RB	NA	B
Containment relative humidity transmitters	SCN	RB	NA	NS
<i>Process Radiation and Environmental Monitoring System, Process Sampling Subsystem</i>				
Non-pressure boundary sampling equipment	SCN	RB, TB, RWB	NA	NS
Pressure boundary sampling equipment (non-contaminated)	SCN	RB, TB, RWB	D	NS
Pressure boundary sampling equipment (contaminated)	SC3	RB, TB, RWB	D	NS
<b>CORE COOLING SYSTEMS</b>				
<b>Isolation Condenser System</b>				
Steam supply, condensate return, standby gas purge piping	SC1	SCCV	A	A
Shutdown Cooling System (SDC) interface piping to containment isolation valve, A and B trains	SC1	SCCV, RB	A	A
Boron Injection System (BIS) interface piping to BIS interface valve, C train	SC1	SCCV	A	A
SDC interface piping from containment isolation valve to downstream redundant isolation valve, A and B trains	SC1	RB	A	A
ICS pools atmospheric vent piping	SC1	RB	B	A
Outer pool to inner pool cross-connect piping	SC1	RB	B	A

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Long-term ICS pool makeup piping (also referred to as flex-makeup piping)	SC3	RB	D	NS
Isolation Condensers (Inside Containment Boundary)	SC1	SCCV, RB	A	A
Isolation Condensers (Outside Containment Boundary)	SC1	RB	B	A
All condensate return valves: Subcomponents supporting pressure boundary	SC1	SCCV	A	A
Open/Close condensate return valves: Subcomponents supporting function to open and remain open	SC1	SCCV	NA	B
Open/Close condensate return valves: Subcomponents supporting function to close and remain closed	SC3	SCCV	NA	NS
Throttling condensate return valves: Subcomponents supporting function to fully open and remain fully open	SC2	SCCV	NA	NS
Throttling condensate return valves: Subcomponents supporting function to throttle, to close, and remain close	SC3	SCCV	NA	NS
Standby gas purge valves: Subcomponents supporting pressure boundary	SC1	SCCV	A	A
Standby gas purge valves: Subcomponents supporting function to close and remain closed	SC1	SCCV	NA	B
Standby gas purge valves: Subcomponents supporting function to open and remain open	SC3	SCCV	NA	NS
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting pressure boundary function	SC1	RB	A	A
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting function to close and remain closed	SC1	RB	NA	B

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting function to open and remain open	SC3	RB	NA	NS
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting pressure boundary function	SC1	RB	A	A
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting function to close and remain closed	SC1	RB	NA	B
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting function to open and remain open	SC3	RB	NA	NS
Outer pool to inner pool cross-connect backflow prevention devices Subcomponents supporting pressure boundary function	SC1	RB	B	A
Outer pool to inner pool cross-connect backflow prevention devices Subcomponents supporting active functions	SC1	RB	NA	B
Flow detection impulse piping and inline passive pressure boundary components	SC1	SCCV, RB	B	A
Flow detection impulse piping excess flow check valve	SC1	RB	B	B
Flow detection differential pressure instrumentation=	SC1	RB	NA	B
Wide range pool level instrumentation used for post-accident monitoring, long term (>72 hours)	SC3	RB	NA	NS
All piping installed temperature instrumentation, pool temperature instrumentation and narrow range pool level instrumentation used for Operating Limits and Conditions monitoring only	SC3	RB, SCCV	NA	NS



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Pneumatic supply tubing and components from the actuator to the control solenoid valves for the open/closed only condensate return valves, containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, SCCV	NA	A
Hydraulic supply tubing and components from the actuator to the control solenoids valves for the throttling condensate return valves	SC2	SCCV	NA	NS
Control solenoid valves for the open/closed only condensate return valves, containment isolation valves, and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, SCCV	NA	B
Control solenoid valves for the throttling condensate return valves	SC2	SCCV	NA	NS
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System to the control solenoid valves for the open/closed only condensate return valves	SC3	SCCV	NA	NS
Pneumatic supply tubing and components from the interface point with Plant Pneumatics to the control solenoid valves for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	NA	NS
Hydraulic supply tubing from the positioner to the control solenoid valves for the throttling condensate return valves	SC3	SCCV	NA	NS
<b>REACTOR SERVICING EQUIPMENT</b>				
<b><i>Refueling Equipment and Servicing</i></b>				
Refueling Platform	SC3	RB	NA	A See Note 8
Fuel Storage Racks	SC3	RB	NA	A See Note 8
Miscellaneous Servicing Equipment	SCN	RB	NA	NS
<b>REACTIVITY CONTROL</b>				
<b><i>Boron Injection System</i></b>				
Injection Pump	SC3	RB	D	NS

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Injection Pump Motor	SC3	RB	NA	NS
Storage Tank	SC3	RB	D	NS
Test Tank	SCN	RB	NAD	NS
Instrumentation – Tank Level, Solution Temperature, Discharge Pressure, Flow Rate	SC3	RB	D	NS
Piping from Tank to Pumps	SC3	RB	D	NS
Piping from Pumps to Outboard Containment Isolation Valve	SC3	RB	D	NS
Injection / Containment Isolation Valves	SC1	RB/SCCV	A	B
Containment Pipe Penetration	SC1	RB/SCCV	A	A
Piping from Containment Penetration to IC return line	SC1	SCCV	A	A
Piping and Valves with no SC function	SCN	RB	D	NS
<b>Control Rod Drive System/High Pressure Injection</b>				
Non-pressure retaining Fine Motor control Rod Drive (FMCRD) scram subcomponents	SC1	SCCV	NA	B
FMCRD RCPB subcomponents except flange ball check valve	SC1	SCCV	A	A
FMCRD Flange Ball Check Valve	SC1	SCCV	A	B
FMCRD Motor	SC2	SCCV	NA	NS
FMCRD separation switches	SC3	SCCV	NA	NS
FMCRD Position Indication Probe with Switches	SC3	SCCV	NA	NS
Hydraulic control unit (HCU) Nitrogen Tank	SC1	RB	B	A
HCU Scram Valve	SC1	RB	B	B
HCU accumulator	SC1	RB	B	B
HCU Scram Solenoid Valve Assembly	SC1	RB	NA	B
HCU Instrument manifold pressure boundary components	SC1	RB	B	A
ARI Valves	SC2	RB	NA	NS
HCU piping and piping between HCU and FMCRD	SC1	RB	B	A
Charging Water piping and valves (except when directly above HCU), pump discharge, drive header, and other piping not part of HCU)	SC3	RB	D	NS
Charging Water Piping and Valves (directly above HCU)	SC3	RB	D	NS

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Purge Water Piping and Valves (except when directly above HCUs)	SC3	RB	D	NS
Purge Water Piping and Valves (directly above HCUs)	SC3	RB	D	NS
Control Rod Drive (CRD) charge pumps	SC3	RB	D	NS
CRD Purge Pumps	SC3	RB	D	NS
CRD Purge FCVs	SC3	RB	D	NS
<b>DECAY HEAT REMOVAL</b>				
<b>ICS Pool Cooling and Cleanup System (ICC)</b>				
Suction Surge Tank Return Guard Pipe	SC1	RB	B	A
All other system piping and components located in RB 1650 Piping (including valves and instrumentation), Pumps/ASDs, HXs, Demineralizer, Dosing Pot	SCN	RB	D	NS
All other components located in ICS pools, including piping, anti-siphon devices, and distribution spargers)	SCN	RB	D	NS
<b>Shutdown Cooling System</b>				
Pump	SC3	RB	D	NS
Heat Exchanger	SC3	RB	D	NS
Leak Detection Equipment supporting Safety Category 1 functions	SC1	RB	C	B
Leak Detection Equipment supporting Safety Category 2 functions	SC2	RB	D	NS
Decay Heat Removal Piping/Valves/ etc.	SC3	RB	D	NS
Overboard Piping/Valves/etc.	SC3	RB/TB	D	NS
<b>Reactor Water Cleanup System</b>				
Heat Exchanger	SC3	TB	D	NS
RB flow element supporting Safety Category 1 and 2 functions	SC1	RB	B	A
RB leak detection instrumentation supporting Safety Category 1 functions	SC1	RB	NA	B
RB leak detection instrumentation supporting Safety Category 2 functions	SC2	RB	NA	B See Note 8
TB flow elements supporting Safety Category 1 and 2 functions	SC1	TB	C	NS See Note 7

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
TB leak detection instrumentation supporting Safety Category 1 functions	SC1	TB	NA	NS See Note 7
TB leak detection instrumentation supporting Safety Category 2 functions	SC2	TB	NA	NS
Piping/Valves/ etc. from RIV to outboard containment isolation valve	SC1	SCCV/RB	B	B
Piping/Valves/ etc. outboard of outer containment isolation valve	SC3	RB/TB	D	NS
Pressure Reduction Station	SC3	TB	D	NS
<b>Fuel Pool Cooling and Cleanup System (FPC)</b>				
General System Piping and Valves	SC3	RB	D	NS
Off-Normal Makeup Piping and Valves	SC3	RB	D	NS
Surge Tanks	SC3	RB	D	NS
Pumps	SC3	RB	D	NS
Filter Elements	SC3	RB	D	NS
Deep Mixed Bed Demineralizers and Service Piping	SC3	RB	D	NS
Heat Exchangers	SC3	RB	D	NS
<b>NUCLEAR FUEL</b>				
Nuclear Fuel Supply	SC1	SCCV, RB	NA	A
<b>RADIOACTIVE WASTE MANAGEMENT SYSTEMS</b>				
<b>Liquid Waste Management System (LWM)</b>				
LWM Equipment	SC3	RB, RWB, TB	D	NS
LWM containment penetration & locked closed isolation valves relied upon for passive pressure integrity in Defense Line 3.	SC1	SCCV	B	A
<b>Solid Waste Management System (SWM)</b>				
SWM Equipment	SC3	RWB	D	NS
Spent Resin Tank	SC3	RWB	D	NS
Sludge Tank	SC3	RWB	D	NS
<b>Offgas System (OGS)</b>				
TB Piping and Valves	SC3	TB	D	NS
Offgas Recombiner	SC3	TB	D	NS
Cooler Condenser	SC3	TB	D	NS
Moisture Separator	SC3	TB	D	NS
Refrigeration Dryers	SC3	TB	D	NS

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Gas Analyzers	SC3	TB	D	NS
RWB Piping and Valves	SC3	RWB	D	NS
Offgas Reheater	SC3	RWB	D	NS
Charcoal Vault / Adsorber Tanks	SC3	RWB	D	NS
Offgas HEPA Filter	SC3	RWB	D	NS
<b>POWER CYCLE SYSTEMS</b>				
<b>Condensate and Feedwater Heating System</b>				
All passive components from the Seismic Restraint near the RB wall to the FW Reactor Isolation Valves	SC1	RB	B	A
Containment isolation valves and system isolation valves for SDC and OLNC.	SC1	RB, SCCV	B	B
Differential Pressure Measurement for Feedwater Leak Detection	SC1	TB	B	NS See Note 7
Components supporting the detection of loss of feedwater	SC1	TB	B	NS See Note 7
System components in the FW flow path from the Condenser interface to the Seismic Restraint near the RB wall	SC3	TB	D	NS
System components in the FW Heater drain path to the condenser	SC3	TB	D	NS
All other system equipment	SCN	TB	D	NS
<b>Condensate Filters and Demineralizers System</b>				
Filters, demineralizers, bypass lines, valves, and related components	SC3	All	D	NS
All other system equipment	SCN	All	D	NS
<b>Main Turbine Equipment</b>				
Main Turbine Equipment and Subsystem	SC3	TB	D	NS
Non-Return Valves	SC3	TB	D	NS
<b>Moisture Separator Reheater System</b>				
Moisture Separator Reheater and associated components supporting drains to Feedwater Heaters	SC3	TB	D	NS
Components supporting steam supply to MSR (Tube and Shell) and the LP Turbines	SC3	TB	D	NS
All other system components	SCN	TB	D	NS
<b>Turbine Bypass System</b>				
Components supporting Turbine bypass	SC3	TB	D	NS

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
All other system equipment	SCN	TB	D or NA	NS
<b>Generator and Exciter</b>				
Generator and Exciter System	SC3	TB	NA	NS
Neutral Grounding Transformer	SCN	TB	NA	NS
Neutral Grounding Resistor	SCN	TB	NA	NS
Automatic Voltage Regulator Cabinet	SC3	TB	NA	NS
Excitation Cabinet	SC3	TB	NA	NS
<b>Main Condenser and Auxiliaries</b>				
Components relied upon for measuring main condenser vacuum (pressure) in support of Defense Line 4a functions.	SC2	TB	D	NS
Components relied upon for measuring main condenser vacuum (pressure) in support of Defense Line 2 functions.	SC3	TB	D	NS
All components associated with: The requirement for MCA to provide the heat sink to condense reactor steam or drainage from the FW heaters and other steam supply users.	SC3	TB	D	NS
All components associated with: The requirement for MCA to provide a means to draw a vacuum and remove non-condensable gases from the condenser shell.	SC3	TB	D	NS
All remaining components not associated with the functions above.	SCN	TB	D	NS
<b>Circulating Water System</b>				
All components associated with: The requirement for CWS to reject heat from the MCA to the environment through the NHS.	SC3	TB, OO	D	NS
All components associated with: The requirement for CWS to reject heat from PCW to the environment through the NHS.	SC3	TB, OO	D	NS
All remaining components not associated with the functions above	SCN	TB, OO	D	NS
<b>STATION AUXILIARY SYSTEMS</b>				
<b>Chilled Water Equipment</b>				
Components supporting HVAC for post-shutdown I&C equipment	SC3	RB, CB	D	NS
Piping and valves inside containment that support containment cooling	SC3	SCCV, RB	D	NS

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment penetration, containment isolation valves, and piping between the CIVs	SC1	SCCV, RB	B	A/B
Air-Cooled chillers, expansion tanks, chiller pumps, and air separators	SC3	RWB	D	NS
Glycol Auto Fill Unit, and Chemical Bypass Unit	SCN	RWB	D	NS
Components support HVAC for non-safety equipment	SCN	ALL	D	NS
<b>Plant Cooling Water System</b>				
Components associated with makeup water supply to the surge tanks and ICS Pools and cleanup heat exchangers.	SCN	ALL	D	NS
All other system equipment	SC3	ALL	D	NS
<b>Plant Pneumatics System</b>				
Containment Penetrations & Isolation Valves	SC1	SCCV, RB	B	A/B
All other system equipment	SC3	ALL	D	NS
<b>Hydrogen Water Chemistry</b>				
All system equipment	SCN	TB	D	NS
<b>Zinc Injection Passivation</b>				
All system equipment	SCN	TB	D	NS
<b>STATION ELECTRICAL SYSTEMS</b>				
<b>SC1 Electrical Distribution System</b>				
All System Equipment	SC1	RB	NA	A
<b>Standby Electrical Distribution System</b>				
SC2 Components	SC2	RB, CB	NA	NS
SC3 Components	SC3	ALL	NA	NS
<b>Normal Electrical Distribution System</b>				
SC3 Components	SC3	ALL	NA	NS
All System Equipment	SCN	ALL	NA	NS
<b>POWER TRANSMISSION SYSTEM</b>				
<b>Switchyard</b>				
All system equipment	SCN	Switchyard	NA	NS
<b>CONTAINMENT AND ENVIRONMENT CONTROL SYSTEMS</b>				
<b>Primary Containment</b>				

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Steel-plate Composite Containment Vessel, including all hatches and seals (such as containment closure head and airlocks) relied upon for passive pressure integrity in Defense Line 3.	SC1	SCCV, RB	B	A
All Containment Penetrations	SC1	SCCV	B	A
Refueling Bellows Seal	TBD	SCCV	TBD	TBD
LRT piping and locked closed containment isolation valves relied upon for passive pressure integrity in Defense Line 3.	SC1	RB	B	A
Passive Containment Cooling System	SC1	SCCV, RB	B	A
PCCS Containment Isolation Valves	SC1	RB	B	B
<b>Containment Inerting System</b>				
Containment Pipe Penetrations	SC1	SCCV	B	A
CIVs, Rupture Disc, Check Valve, and Associated Piping	SC1	RB	B	A/B
Sparger Piping	SC3	RB	D	NS
All other system equipment and piping	SC3	RG, RWB, OO	D	NS
<b>Containment Cooling System</b>				
Drain valves	SCN	SCCV	D	NS
All other system equipment	SC3	SCCV	D	NS
<b>STRUCTURE AND SERVICING SYSTEMS</b>				
<b>Cranes, Hoists, and Elevators</b>				
All system equipment	SCN	ALL	NA	NS
<b>Heating Ventilation and Cooling System</b>				
MCR Emergency HVAC	TBD	CB	NA	NS
SCR Emergency HVAC	TBD	RB	NA	NS
RB DCIS Rooms and SCR Fan Coil Units (FCU)	TBD	RB	NA	NS
Defense Line 2 FCUs	SC3	CB	NA	NS
Defense Line 4a FCUs	SC2	CB	NA	NS
RB Refueling Floor Isolation Dampers	SC3	RB	NA	NS
All other system equipment	SCN	ALL	NA	NS
<b>Fire Protection System (FPS)</b>				
System components that support DL2 or DL4b functions (Piping, valves and sprinklers)	SC3	ALL	D	TBD



NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
All other system equipment	SCN	RB and CB	D	TBD
<b><i>Equipment and Floor Drain System</i></b>				
Piping and valves and supports forming part of the containment boundary	SC1	RB	B	A/B
Drain piping and valves, including supports.	SC3	ALL	D	NS
All other general equipment and floor drain system equipment	SC3	ALL	D or NA	NS
Oily waste sump system and other non-radioactive subsystems	SCN	TB	NA	NS
<b><i>Water, Gas, and Chemical Pads</i></b>				
Components required to provide standby diesel fuel oil storage and transfer	SC3	ALL	D	NS
All other system equipment	SCN	ALL	D	NS

NOTES:

1. SC determination and methodology is discussed in Subsections 3.2.1 and 3.2.2.
2. Location Codes:
  - a. SCCV: Containment Vessel
  - b. RB: Reactor Building
  - c. TB: Turbine Building
  - d. CB: Control Building
  - e. RWB: Radwaste Building
  - f. OO: Outdoors On-site
  - g. OL: Any Other Location
  - h. ALL: All locations
3. Quality group classifications is discussed in Subsection 3.2.4.
4. Seismic categories are discussed in Subsection 3.2.3. Any items classified as NS are subject to evaluations for Seismic Interaction as discussed in Subsection 3.2.3.1.
5. Structures, systems and components required to be designed in accordance with Radioactive Waste Management requirements from RG 1.143 for Category RW-IIa, shall meet the guidance of NRC Regulatory Guide 1.143, as applied to radioactive waste management systems, with regard to quality, seismic, and quality group requirements.
6. Other abbreviations.
  - a. TBD: To Be Determined – classification information is to be provided later in the BWRX-300 design process
  - b. NA: Not Applicable
7. Components classified as SC1 may be assigned to a Seismic Class lower than A or B provided they are of a fail-safe design such that the failure of those component(s) does not adversely affect the ability to achieve the safety function.
8. Although these components are not SC1, they are seismically qualified because they are credited with monitoring leakage of reactor coolant under the scope of Regulatory Guide 1.45 or are related to handling and storage of used nuclear fuel.

## **APPENDIX 3B – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES**

### **3.13 Introduction**

This appendix describes the major computer programs used in the analysis and design of the BWRX-300 Seismic Category structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3.12-18) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3.12-12).

GEH maintains an ISO 9001:2015, "Quality Management Systems - Requirements," International Organization for Standardization" (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12, "Management System Requirements for Nuclear Facilities" (Reference 3.12-6).

#### **3.13.1 ACS SASSI v4**

**Description:** ACS SASSI is a finite element computer code on the Microsoft Windows PC platforms for performing 3D dynamic soil-structure interaction (SSI) analysis to analyze the effect of seismic ground motions on structures. The analysis is performed in the frequency domain using linear or equivalent-linear material properties for the structure and soil.

**Validation:** The software is approved for production use under GEH procedure on engineering software for design and analysis software.

**Extent of Application:** ACS SASSI is used to perform seismic and static SSI and structure-soil-structure-interaction (SSSI) analyses, as applicable.

#### **3.13.2 ANSYS v17**

**Description:** ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

**Validation:** The software is approved for production use under GEH procedure on engineering software for design and analysis software.

**Extent of Application:** This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

### **3.13.3 Ansys LS-Dyna v2021**

**Description:** Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

**Validation:** This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

**Extent of Application:** Ansys LS-DYNA is used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

### **3.13.4 SSI-StressCoord v1**

**Description:** The STRESS\_POST program is an auxiliary program to post-process the ACS SASSI NQA V4 STRESS result binary database. The STRESS\_POST program includes an ensemble of STRESS database processing functionalities which were customized for the GEH engineers for application to the BWRX-300 SMR seismic SSI analysis projects. The STRESS\_POST customized program is based on specific implementations incorporated in the ACS SASSI NQA V4 User Interface (UI) capabilities, such as the CTVEC and the CTCCV commands, and existing STRESS binary database verification tools used in-house during the development over years of the STRESS module.

**Validation:** The software is approved for production use under GEH procedure on engineering software for design and analysis software.

**Extent of Application:** This STRESS\_POST Program is used for post-processing the ACS SASSI STRESS binary databases for Integrated RB Walls and Floors in batch mode.

### **3.13.5 GT STRUDL**

#### **3.13.5.1 Description**

GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

#### **3.13.5.2 Validation**

This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

#### **3.13.5.3 Extent of Application**

GT STRUDL is used to for the structural analysis and design of non-Seismic Category A structures.

## **APPENDIX 3C – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF MECHANICAL STRUCTURES, SYSTEMS AND COMPONENTS**

### **3.14 Introduction**

As discussed in Subsection 3.6.1.1, this appendix describes the major computer programs used in the analysis of mechanical SSC.. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

#### **3.14.1 ANSYS v17**

**Description:** ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

**Validation:** The software is approved for production use under GEH procedure on engineering software for design and analysis software.

**Extent of Application:** This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

#### **3.14.2 PBLE v1**

**Description:** Steam Dryer Analysis

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** PBLE calculates the acoustic loads on a steam dryer based on measurements of pressure along the main steam lines or pressures measured directly on the face of the steam dryer. The loads are then used in a finite element model to calculate the stresses in the dryer.

#### **3.14.3 SIMCENTER 3D Acoustics v2022**

**Description:** Used for modeling dryer acoustic loads and instrumentation diagnostics. Simcenter 3D is a unified, scalable, open and extensible environment for 3D CAE with connections to design,

1D simulation, test, and data management. Fast and accurate solvers power structural, acoustics, flow, thermal, motion, and composites analyses, as well as optimization and multi-physics simulation.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent Of Application:** SIMCENTER Finite elements acoustic software will be used to model and calculate acoustic wave propagation in fluid (steam, water) mediums.

#### 3.14.4 GT STRUDL

**Description:** GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** GT STRUDL will be used to perform structural analysis and qualification of supports.

#### 3.14.5 HyperMesh

**Description:** HyperMesh is the market-leading, multi-disciplinary finite element pre-processor which manages the generation of the largest, most complex models, starting with the import of a CAD geometry to exporting a ready-to-run solver file. With its advanced geometry and meshing capabilities, HyperMesh provides an environment for rapid model generation.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** HyperMesh is a tool which will be used to generate mechanical models for complicated mechanical components. This tool will serve as a pre-processor to build mesh models, no calculations get performed with Hypermesh.

#### 3.14.6 ERSIN v3

**Description:** Piping Analysis Software. Secondary Response Spectra for control panels, equipment racks, etc.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** ERSIN is used to generate secondary response spectra for pipe and floor mounted equipment. Example applications include control panels, equipment racks, Main Steam Isolation Valves (MSIVs), Safety Relief Valves (SRVs), Hydraulic Control Units (HCUs), et cetera. ERSIN03P software has three input options: 1) card decks, 2) SAP software decks, and 3) PISYS software decks. ERSIN03P can be used with SAP version 4G07P (Ref. 5-1) and PISYS version 08P (Ref. 5-2) structure/piping models only. If a card input is used, a mass normalized mode shape is required. ERSIN03P is not applicable for axisymmetric analyses using a Fourier Decomposition technique.

#### 3.14.7 RINEX Computer Program

**Description:** RINEX is a computer code used to interpolate and extrapolate amplified response spectra used in the response spectrum method of dynamic analysis. RINEX is also used to generate response spectra with non-constant model damping. The non-constant model damping

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

analysis option can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13" (Reference 3.12-20) damping values.

**Validation:** Hand calculations and test cases analyzed are used to demonstrate the program's applicability and validity.

**Extent of Application:** This program is used to generate multiple damping spectra for piping.

### 3.14.8 PDA (Civil)

**Description:** Pipe Dynamic Analysis (PDA) Pipe Whip Restraint Analysis

**Validation:** This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

**Extent of Application:** GEH in-house program for calculating pipe whip response under postulated break conditions. Determines response for a standard configuration which utilizes U-type pipe whip restraint.

### 3.14.9 PIPESTRESS

**Description:** PIPESTRESS (developed under a Quality Assurance Program compliant with the ASME NQA-1 (Reference 3.12-18) standard along with 10 CFR 21, "Reporting of Defects and Noncompliance" (Reference 3.12-9) and 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" (Reference 3.12-10)) is a pipe stress and flexibility analysis program, used for the evaluation of structural response and stress levels of piping systems against the requirements of industry codes and standards.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The plant layout, isometric drawings, P&ID, PFD, etc. will be used to build the piping model in PIPESTRESS, then PIPESTRESS will calculate the displacement, force/moment and stress. This software has the piping information, pipe routing & system information for BWRX-300 & some equipment information.

### 3.14.10 FLOMASTER v2021.1

**Description:** Uses simulation to offer reliable & accurate solvers and solutions for fluids engineering

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** Simcenter Flomaster is a unique thermo-fluid system simulation software tool used to simulate thermo-fluid systems; facilitating upfront engineering to reduce cost and lead times in product development and maintenance. It has an extensive library of component models, pre-populated with reliable performance data, Flomaster allows fluid system design to start before CAD data is available and component suppliers have been selected.

### 3.14.11 Ansys LS-Dyna v2021

**Description:** Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all

the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** Ansys LS-DYNA will be used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

#### **3.14.12      3KeyMaster v2021 (ICE/Plant Integration Engineering/Systems Engineering)**

Description: Plant-wide physics-based simulation supporting engineering design options, confirmation, and future reactor operator training full scope simulator (FSS) in accordance with ANS Std 3.5.

Validation: This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

Extent of Application: 3KeyMaster is used to generate plant layout schematics & run test simulations for new plant setups through variable/parameter manipulation for OPG.

## **APPENDIX 3D – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF ELECTRICAL STRUCTURES, SYSTEMS AND COMPONENTS**

### **3.15 Introduction**

This appendix describes the major computer programs used in the analysis of electrical SSC. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

#### **3.15.1 ETAP v2021.1 (ICE Systems/I&C Tech)**

**Description:** Electrical Transient Analyzer Program (ETAP) is an electrical network modeling and simulation software tool used by power systems engineers to create an "electrical digital twin" and analyze electrical power system dynamics, transients and protection.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** ETAP is the Global Market and Technology Leader of power systems solutions for a broad spectrum of sectors including Generation, Transmission, Distribution, Transportation, Industrial, and Commercial. The most comprehensive and integrated model-driven solutions for design, simulation, analysis, optimization, monitoring, operation, and automation of electrical power systems.

#### **3.15.2 LDRA (I&C Tech/ICE Systems)**

**Description:** Liverpool Data Research Associates is a provider of software analysis, and test and requirements traceability tools for the Public and Private sectors and a pioneer in static and dynamic software analysis.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** LDRA is a tool used to perform unit/module testing on software functions and components. It allows us to create and store test cases so we can perform regression testing, and it also allows us to execute the test cases on the target hardware (in this case an ARM Cortex-A9 processor).

#### **3.15.3 Quartus II (I&C Tech/ICE Systems)**

**Description:** Tools that provide FPGA compiler, simulation, and programming capabilities.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.



**Extent of Application:** Quartus is a tool used to develop applications for programmable logic devices such as PLDs and FPGAs. Applications in this case means the logic that the device implements. For example, it could be logic that provides a 2 out of 3 votes, it could be something that processes digital communications such as our fibre optic links, etc. Included in the software is something called timing analysis, which is a methodology for ensuring the logic inside the device meets timing characteristics. It also includes support for a simulator. The simulator allows engineers to evaluate the functionality of their logic by specifying input and examining how the logic reacts (e.g., verify the correctness of the design). The simulator does not require a physical device.

## **APPENDIX 3E – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSES STRUCTURES, SYSTEMS AND COMPONENTS – NUCLEAR FUELS**

### **3.16 Introduction**

This appendix describes the major computer programs used in the analysis of nuclear fuels. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

#### **3.16.1 EPRI: Acube v11**

**Description:** Advanced cutset upper bound estimator

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use ACUBE to post-process result cutsets using a Binary Decision Diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cutsets and returns an estimate of the probability for a given top event using the BDD method. The BDD method is more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

#### **3.16.2 EPRI: CAFTA v11**

**Description:** CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The CAFTA software will be qualified to complete all designed functions within the software. The use of the CAFTA software will be acceptable for use as is. Note that the testing will not cover every possible variation or combination of use for the software but it will validate the software operates as intended for within the standard operating configuration of the software.

### 3.16.3 EPRI: MAAP v5

**Description:** The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licenced computer software - is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Assessment (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use MAAP to analyze reactor thermal-hydraulic and containment response to transients as well as severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

### 3.16.4 EPRI: PRAQuant v11

**Description:** Accident Sequence Quantification. In performing a fault tree based analysis it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use PRAQuant in the processing of the combined hazard model to generate a combined hazard cutset output. PRAQuant is a processing software to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation, etc.) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification engine generates an output cutset file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

### 3.16.5 FURST (Core & Fuel)

**Description:** Static & dynamic modeling

**Validation:** The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

**Extent of Application:** Mechanical design of core internals loads, deflections, and stress analysis for X300

### 3.16.6 GTRAC v1

**Description:** Post-processing TRACG graphics file to edit desired output

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

### 3.16.7 MACCS v4

**Description:** The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (USNRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** MACCS will be used as part of the licensing basis events analysis in radiological consequences.

### 3.16.8 MCNPX v6

**Description:** Monte Carlo N-Particle Transport is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies and is developed by Los Alamos National Laboratory.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** MCNP will be used for performing criticality and shielding analyses. MCNP can be used in several transport modes: neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10<sup>-11</sup> MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes, the photon energy regime is from 1 keV to 100 GeV, and the electron energy regime is from 1 KeV to 1 GeV. The capability to calculate keff eigenvalues for fissile systems is also a standard feature.

### 3.16.9 ORIGEN v1

**Description:** ORIGEN is a one-group depletion and radioactive decay computer code. ORIGEN is used to calculate the radionuclide composition and other related properties of nuclear materials (irradiated fuel isotope inventory).

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** ORIGEN is used for calculating core inventories of isotopes, and sometime for performing activation analyses of various materials or components.

### **3.16.10 PANAC v11**

**Description:** PANAC (PANACEA) is the computer program used for the detailed nuclear calculations of the BWR Core. It is a steady-state, three-dimensional, one and one half energy group, diffusion theory computer program with coupled nuclear and thermal-hydraulic representation of the reactor Core.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** The BWR Core Simulator (PANAC11A/P) is a steady-state, three-dimensional coupled nuclear-thermalhydraulic computer program representing a BWR core. An automated plant heat balance option is used for modeling of the external flow loop. Provisions are made for fuel cycle and thermal limits calculations. The program is used for detailed three-dimensional design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A special power exposure iteration option is available for target exposure distribution and cycle length predictions. PANAC11A/P includes the effect of Doppler broadening as a function of moderator density, exposure, control and moderator density history for a given fuel type. The nuclear model is based on coarse-mesh nodal, improved 1-1/2 group (quasi-two group), static diffusion theory. The diffusion equations are solved using the fast energy group. Resonance energy neutronic effects are included in the model by relating the resonance fluxes to the fast energy flux. The thermal flux is represented by an asymptotic expansion using a slowing down source from the epithermal region. A spectral history reactivity model and control blade history reactivity model are included. Control blade history local peaking effects are also incorporated in the nuclear model. A pin power reconstruction model is implemented to account for the effect of flux gradients across the nodes on the local peaking distribution. Neutronic parameters used by PANAC11A/P are obtained from the two-dimensional lattice physics code (TGBLA06) and parametrically fitted as a function of moderator density, exposure, control and moderator density history for a given fuel type.

### **3.16.11 PRIME v3**

**Description:** The PRIME03P computer program is used to calculate the thermal/mechanical response of nuclear fuel to time varying power histories.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** PRIME03P is used for steady-state and transient licensing analysis of UO<sub>2</sub> and (U,Gd)O<sub>2</sub> fuel with (and without) additive material. PRIME03P is used for steady-state and transient licensing analysis as well as qualification cases of Recrystallized Annealed Zircaloy-2 cladding. Additionally, PRIME03P may be used with Stress-Relieved Annealed Zircaloy-4 cladding of either 70 % or 30 % cold work for qualification cases, but not for licensing analysis.

### **3.16.12 RAMP: GALE v3.2**

**Description:** The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as "inputs"; GALE asks the

operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

#### **3.16.13 RAMP: HABIT v2.2**

**Description:** HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

#### **3.16.14 RAMP: DandD v2.1**

**Description:** A code for screening analyses for licence termination and decommissioning.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning" (Reference 3.12-15).

#### **3.16.15 RAMP: GENII v2.10**

**Description:** GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

#### **3.16.16 RAMP: MILDOS v4**

**Description:** Radiological dose commitment calculation code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

**3.16.17 RAMP: NRC-RADTRAN v6.02.1**

**Description:** Risk & Consequence analysis code

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The USNRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

**3.16.18 RAMP: PIMAL v4.1.0**

**Description:** GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

**3.16.19 RAMP: TurboFRMAC v2021 11.0.2**

**Description:** Radiological Hazard evaluation code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

**3.16.20 RAMP: VARSKIN v1.0**

**Description:** Occupational Dose Analysis Code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c), "Occupational dose limits for Adults" (Reference 3.12-11), which states that the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm<sup>2</sup> of skin at a tissue depth of 0.007 centimeters (7 mg/cm<sup>2</sup>).

#### **3.16.21      SAP4G07P v7**

**Description:** SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN and has been compiled and run on Windows 7 (32 bit), Windows 7 (64 bit), and Windows 2003 and 2012 servers.

#### **3.16.22      SCALE v6**

**Description:** A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

#### **3.16.23      TGBLA v6**

**Description:** LANCR will replace TGBLA. Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behavior of the fuel in the core

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** TGBLA06 is a lattice design computer program for conventional BWRs, which have the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods and approximations for centered and offset water boxes, may be introduced into cells of the 2D mesh, which TGBLA06 solves. The 8x8 lattice can have up to four cells per water rod; the 9x9 lattice can have up to 3.5 cells per water rod; the 10x10 lattice can have up to four cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water cross designs such as 8x8 and 10x10 water cross lattices, are qualified. TGBLA06 is qualified for water box designs where the water box is simulated by the use of nine water rods. Although TGBLA06 is capable of analyzing 11x11 and 12x12 lattices, MOX fuel and other design configurations, it has not been qualified for them. TGBLA06 solves 2D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenized cross sections. Also, TGBLA06 performs burnup calculations for generating input to the BWR 3D simulator. In addition, TGBLA06 generates the rod-by-rod neutron cross sections, gamma smeared power distributions and flux discontinuity factors. The ring-by-ring gamma source distribution in gadolinium rods is not correct and should not be used.



#### **3.16.24 TRACG v4**

**Description:** TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

#### **3.16.25 SEISM v5**

**Description:** The SEISM program can be used for the non-linear response prediction of structural system with spring, damper, friction & stop element, under dynamic loads. The program employs the component element method and can account for impact and friction forces effect. SEISM program performs calculations in double precision.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** SEISM can be used for the non-linear time history response prediction of structural systems with spring, damper, friction and stop elements under dynamic loads. The program employs the component element method and can account for impact and friction force effects. When running SEISM, the user can select to run any of its four modules (CRTFI, SEPRE, SEISM, SEPST) individually or combined within a single session. Output of one module may be passed to and used as input to the next module.

#### **3.16.26 DECAY v1**

**Description:** DECAY01A calculates the decay heat power fraction after certain operation period and exposure of a fissile core.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** DECAY01A is an Engineering Computer Code developed by GE Hitachi Nuclear Energy (GEH) as a method to determine the decay heat (shutdown power) for BWR fuel. The code was created in response to USNRC IN96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly" (Reference 3.12-16) that brought attention to the extreme variation in decay heat calculations throughout the country. This was due to either overly conservative assumptions or a misapplication of the ANS Decay Heat Standards. The DECAY01A code has therefore gone to great lengths to assure both the validity and applicability of its calculations. DECAY01A works as a function of both the ANSI/ANS-5.1-1979, "American National Standard Decay Heat Power in Light Water Reactors" (Reference 3.12-17) or ANSI/ANS-5.1-1994 (Reference 3.12-17) decay heat standards used for domestic and advanced reactor

designs respectively. These standards set forth values of decay heat from fission products of  $^{235}\text{U}$ ,  $^{239}\text{Pu}$ ,  $^{238}\text{U}$  and  $^{241}\text{Pu}$ ; and decay heat from actinides  $^{239}\text{U}$  and  $^{239}\text{Np}$ . DECAY01A also includes the decay heat contribution from other Actinides (in addition to  $^{239}\text{U}$  and  $^{239}\text{Np}$  which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

### **3.16.27 GTRAC v1**

**Description:** Post-processing TRACG graphics file to edit desired output

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

## **APPENDIX 3F – COMPUTER PROGRAMS USED IN ENVIRONMENTAL AND RADIOLOGICAL ANALYSES SUPPORTING THE DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS**

### **3.17 Introduction**

This appendix describes the major computer programs used in deterministic and probabilistic safety analyses. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

#### **3.17.1 ADDAM Version 1.4.2**

**Description:** The ADDAM (Atmospheric Dispersion and Dose Analysis Method) computer code computes the statistical distribution of radiation doses to an individual or population after the airborne release of radioactive material into the environment. See Chapter 15, Subsection 15.5.1.2.5 for a description.

#### **Validation**

Validation of this tool is in compliance with the OPG project quality plan.

#### **Extent of Application**

See Chapter 15, Subsection 15.5. for extent of application.

#### **3.17.2 DECAY v1**

**Description:** DECAY01A calculates the decay heat power fraction after certain operation period and exposure of a fissile core.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** DECAY01A is an Engineering Computer Code developed by GE Hitachi Nuclear Energy (GEH) as a method to determine the decay heat (shutdown power) for BWR fuel. The code was created in response to USNRC IN96-39 (Reference 3.12-16) that brought attention to the extreme variation in decay heat calculations throughout the country. This was due to either overly conservative assumptions or a misapplication of the ANS Decay Heat Standards. The DECAY01A code has therefore gone to great lengths to assure both the validity and applicability of its calculations. DECAY01A works as a function of both the ANSI/ANS-5.1-1979 (Reference

3.12-17) or ANSI/ANS-5.1-1994 (Reference 3.12-17) decay heat standards used for domestic and advanced reactor designs respectively. These standards set forth values of decay heat from fission products of <sup>235</sup>U, <sup>239</sup>Pu, <sup>238</sup>U and <sup>241</sup>Pu; and decay heat from actinides <sup>239</sup>U and <sup>239</sup>Np. DECAY01A also includes the decay heat contribution from other Actinides (in addition to <sup>239</sup>U and <sup>239</sup>Np which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

### **3.17.3 RADTRAD (Analytical Methods/ Radiological Analysis)**

**Description:** RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation.

**Validation:** The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

**Extent of Application:** The RADTRAD code is used for calculating accident doses, calculating transport of fission products inside the plant after an accident, performing filter loading calculations for post-accident.

### **3.17.4 RAMP: GALE v3.2**

**Description:** The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as “inputs”; GALE asks the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

### **3.17.5 RAMP: HABIT v2.2**

**Description:** HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

### **3.17.6 RAMP: DandD v2.1**

**Description:** A code for screening analyses for licence termination and decommissioning.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and

justifications of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512 (Reference 3.12-15).

#### **3.17.7 RAMP: GENII v2.10 (Analytical Methods/Radiological Analysis)**

**Description:** GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

#### **3.17.8 RAMP: MILDOS v4**

**Description:** Radiological dose commitment calculation code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

#### **3.17.9 RAMP: NRC-RADTRAN v6.02.1**

**Description:** Risk & Consequence analysis code

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

#### **3.17.10 RAMP: PIMAL v4.1.0**

**Description:** GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms

and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

### **3.17.11 RAMP: TurboFRMAC v2021 11.0.2**

**Description:** Radiological Hazard evaluation code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

### **3.17.12 RAMP: VARSKIN v1.0**

**Description:** Occupational Dose Analysis Code

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c) {Reference 3.12-11), which states that the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm<sup>2</sup> of skin at a tissue depth of 0.007 centimeters (7 mg/cm<sup>2</sup>).

### **3.17.13 SAP4G07P v7**

**Description:** SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN.

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN and has been compiled and run on Windows 7 (32 bit), Windows 7 (64 bit), and Windows 2003 and 2012 servers.

### **3.17.14 SCALE v6**

**Description:** A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

### 3.17.15 TGBLA v6

**Description:** LANCR will replace TGBLA. Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behavior of the fuel in the core

**Validation:** Validation of this tool is in compliance with the OPG project quality plan.

**Extent of Application:** TGBLA06 is a lattice design computer program for conventional BWRs, which have the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods and approximations for centered and offset water boxes, may be introduced into cells of the 2D mesh, which TGBLA06 solves. The 8x8 lattice can have up to four cells per water rod; the 9x9 lattice can have up to 3.5 cells per water rod; the 10x10 lattice can have up to four cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water cross designs such as 8x8 and 10x10 water cross lattices, are qualified. TGBLA06 is qualified for water box designs where the water box is simulated by the use of nine water rods. Although TGBLA06 is capable of analyzing 11x11 and 12x12 lattices, MOX fuel and other design configurations, it has not been qualified for them. TGBLA06 solves 2D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenized cross sections. Also, TGBLA06 performs burnup calculations for generating input to the BWR 3D simulator. In addition, TGBLA06 generates the rod-by-rod neutron cross sections, gamma smeared power distributions and flux discontinuity factors. The ring-by-ring gamma source distribution in gadolinium rods is not correct and should not be used.

### 3.17.16 TRACG v4

**Description:** TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

### 3.17.17      **IMPACT**

**Description:** IMPACT is a customizable tool that allows the user to assess the transport and fate of contaminants through a user-specified environment.

**Validation:** The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

**Extent of Application:** IMPACT performs the calculations for CSA N288.1:14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities", R2019 (Reference 3.12-14). The code calculates the doses from routine effluent emission from a plant that are the results of normal operation.



## **APPENDIX 3G – COMPUTER PROGRAMS USED IN THE DESIGN OF COMPONENTS, SYSTEMS AND STRUCTURES IN SAFETY ANALYSES (PRA AND DETERMINISTIC)**

### **3.18 Introduction**

This appendix describes the major computer programs used in the analysis of the safety-related components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

#### **3.18.1 EPRI: Acube v11**

**Description:** Advanced cutset upper bound estimator

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use ACUBE to post-process result cutsets using a Binary Decision Diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cutsets and returns an estimate of the probability for a given top event using the BDD method. The BDD method is more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

#### **3.18.2 EPRI: CAFTA v11**

**Description:** CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The CAFTA software will be qualified to complete all designed functions within the software. The use of the CAFTA software will be acceptable for use as is. Note that the testing will not cover every possible variation or combination of use for the software but it will validate the software operates as intended for within the standard operating configuration of the software.

### 3.18.3 EPRI: FRANX v11

**Description:** Development of PRA Hazards models (Fire, Flood, High Winds, Seismic, etc.)

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use FRANX in the development of the Internal Fire, Internal Flood, Seismic, and High Winds hazard analyses. Specifically, FRANX will be used to build hazard specific scenarios and generate one-top models for later combination into an integrated hazard model. The FRANX software is a tool for analyzing external event risk. This tool is used to manage and develop the scenarios, calculate the probabilistic impact on core damage, and generate one-top solution models.

### 3.18.4 EPRI: FTREx v1.8

**Description:** FTREX reads a fault tree that consists of Boolean equations for system failure and generates cut sets that are minimal combinations of component failures that cause system failure.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** This software will have all functionality qualified and be valid for use with the necessary interfacing software (e.g., FRANX, CAFTA, PRAQuant) or independently of those software. The software must be accessible from the interfacing software locations as well as have permission to read and write files to a temp directory and a defined output file directory.

### 3.18.5 EPRI: HRA Calculator

**Description:** Supports development of PRA Human Reliability Analyses

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use the HRA Calculator to develop the human reliability analysis, calculate the human error probabilities, and develop a dependency analysis for the credited operator actions. The HRA Calculator provides a step by step process for developing the HRA applying one of the following methods: CBDTM, HCR/ORE, ASEP, SPAR-H, THERP.

### 3.18.6 EPRI: MAAP v5

**Description:** The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licenced computer software - is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Assessment (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use MAAP to analyze reactor thermal-hydraulic and containment response to transients as well as severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

### 3.18.7 EPRI: PRAQuant v11

**Description:** Accident Sequence Quantification. In performing a fault tree based analysis it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use PRAQuant in the processing of the combined hazard model to generate a combined hazard cutset output. PRAQuant is a processing software to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation, etc.) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification engine generates an output cutset file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

### 3.18.8 ActivePoint HMI/CIMPLICITY 11

**Description:** Digital user interface design and display software by GE Power that runs using GE Digital CIMPLICITY HMI/SCADA automation platform.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The HFE team is using the software to design the BWRX-300 digital user interfaces. The scope of the interfaces is all display screens run by the DCIS, and any other platforms that can communicate directly with CIMPLICITY.

### 3.18.9 Control ST – ToolboxST Tool

**Description:** GE Power's ControlST\* software suite provides the foundation for the Mark\* V1e Control System in a wide range of applications, including control, safety integrity level, monitoring, and protection of assets. ToolboxST is one of the tools within ControlST, used for process configuration and diagnostics software for process, SIL, excitation and power conversion

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** For BWRX-300, the HFE team is using ToolboxST to provide early dynamic features and testing capability for the digital user interfaces designed using ActivePoint HMI/CIMPLICITY. The tool allows emulation of "live" screen features without the need for a plant simulation model driving the software. This allows early usability testing of digital user interfaces, as part of the HFE design testing and evaluation set of activities. The software is not used in production.

### 3.18.10 EPRI: SysImp v11

**Description:** Analysis of PRA Importance Measures. SysImp is a software tool used to calculate the importance of basic events, or collections of those events, in a risk model. SysImp is designed for risk models where components, equipment trains, and systems are represented by groups of basic events.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use SysImp to preform risk importance sensitivities, calculations, and grouping system importance. SysImp allows for deriving insights from risk importance rankings, estimating total plant risk given a specific change, and collective risk importance measures.

### **3.18.11      EPRI: UNCERT v11**

**Description:** PRA Uncertainty Propagation analysis tool. Uncertainty Evaluation Tool (UNCERT). UNCERT can read the cut set or sequence data created from CAFTA and calculate the uncertainty of the cut set result.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project will use UNCERT to perform the parametric uncertainty calculations on the output cut sets. The UNCERT software will take a defined input (e.g., cut set file and associated CAFTA RR database) and perform the uncertainty analysis utilizing either a Monte Carlo or Latin Hypercube sampling method. The output will calculate the metrics for the cut set using that defined method.

### **3.18.12      GOTHIC v8**

**Description:** GOTHIC is a procured software from Zachry Nuclear Engineering, Inc. for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** GOTHIC is used to perform a sensitivity analysis for the passive containment cooling system while developing the design.

### **3.18.13      MACCS v4**

**Description:** The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (NRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** MACCS will be used as part of the licensing basis events analysis in radiological consequences.

### **3.18.14      RAMP: NRC-RADTRAN v6.02.1**

**Description:** Risk & Consequence analysis code

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

### **3.18.15      SCALE v6**

**Description:** A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

### **3.18.16      TRACG v4**

**Description:** TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

### **3.18.17      VTR.LMP**

**Description:** Package of functions and data frames supporting VTR LMP applications. This package was developed using open-source code R. Currently only functions on a Mac platform.

**Validation:** The software qualification process is being followed and verification and validation is in progress.

**Extent of Application:** The BWRX-300 project currently does not use this code package; however, developmental work is in progress to explore the application of this software to BWRX-

NEDO-33952 REVISION 1  
NON-PROPRIETARY INFORMATION

300. The VTR.LMP R code package contains the processing commands necessary for gathering the inputs and running them through the LMP code package functions. The final licensing basis events are processed in this code package for use with the Frequency-Consequence plot.

Note: There is a developmental X300.LMP that would be the starting point for future applications of this code package.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33953

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 4  
Reactor**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.



NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CST	Condensate Storage Tank
DBA	Design Basis Accident
DEC	Design Extension Conditions
DL	Defense Line
DPS	Diverse Protection System
DSA	Deterministic Safety Analysis
ERICP	Emergency Rod Insertion Control Panel
ESBWR	Economic Simplified Boiling Water Reactor
FMCRD	Fine Motion Control Rod Drive
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
ISI	In-Service Inspection
IST	In-Service Testing
KKM	Kernkraftwerk Mühleberg (BWR/4 in Switzerland)
LHGR	Linear Heat Generation Rate
LPRM	Local Power Range Monitor
MFLCPR	Maximum Fraction Limiting Critical Power Ratio
MCPR	Minimum Critical Power Ratio
MFLPD	Maximum Fraction Limiting Power Density

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
MLHGR	Maximum Linear Heat Generation Rate
NBS	Nuclear Boiler System
OLMCPR	Operating Limit Minimum Critical Power Ratio
PA	Postulated Accident
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring System
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SCRRRI	Selected Control Rod Rapid Insertion
SDC	Shutdown Cooling System
TRACG	Transient Reactor Analysis Code General Electric
USNRC	U.S. Nuclear Regulatory Commission
$\Delta$ CPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

## TABLE OF CONTENTS

<b>4.0</b>	<b>REACTOR.....</b>	<b>4-1</b>
4.1	Summary Description .....	4-1
4.1.1	Reactor Pressure Vessel .....	4-1
4.1.2	Reactor Internal Components .....	4-2
4.1.3	Reactor Core .....	4-2
4.1.4	Fuel Assembly and Control Rod Assembly .....	4-2
4.1.5	Nuclear Instrumentation.....	4-3
4.1.6	Analysis Techniques.....	4-3
4.1.7	References .....	4-3
4.2	Fuel System Design .....	4-8
4.2.1	Design Bases .....	4-8
4.2.2	Fuel Assembly and Control Rods Description .....	4-10
4.2.3	Fuel Design Evaluation .....	4-14
4.2.4	Control Rods Design Evaluation .....	4-17
4.2.5	Inspection and Testing.....	4-20
4.2.6	References .....	4-23
4.3	Nuclear Design .....	4-27
4.3.1	Design Bases .....	4-27
4.3.2	Core Nuclear Design Description.....	4-28
4.3.3	Core Nuclear Analytical Methods.....	4-30
4.3.4	Core Nuclear Design Evaluation .....	4-31
4.3.5	Changes from Previous Reactor Design .....	4-34
4.3.6	References .....	4-34
4.4	Thermal and Hydraulic Design Basis .....	4-46
4.4.1	Design Bases .....	4-46
4.4.2	Thermal and Hydraulic Methods .....	4-48
4.4.3	Description of the Thermal and Hydraulic Design of the Reactor Core .....	4-50
4.4.4	Thermal and Hydraulic Evaluation .....	4-52
4.4.5	Evaluation of the Validity of Thermal and Hydraulic Design Techniques.....	4-53
4.4.6	References .....	4-54
4.5	Reactor Internals Material.....	4-59

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

4.5.1	References .....	4-59
4.6	Design of Reactivity Control Systems .....	4-60
4.6.1	Control Rod Drive Design Bases .....	4-60
4.6.2	Description .....	4-60
4.6.3	Safety Evaluation of the CRD System.....	4-70
4.6.4	Testing, Inspection, and Maintenance.....	4-71
4.6.5	References .....	4-72
4.7	Core Monitoring .....	4-85
4.7.1	System and Equipment Functions .....	4-85
4.7.2	References .....	4-87
4.8	Thermal-Hydraulic Stability .....	4-88
4.8.1	Background .....	4-88
4.8.2	Design Bases .....	4-88
4.8.3	Types of BWR Oscillations .....	4-88
4.8.4	Design Criteria.....	4-89
4.8.5	Stability Solution Design .....	4-89
4.8.6	Stability Analysis Methods .....	4-90
4.8.7	Stability Evaluation .....	4-91
4.8.8	References .....	4-91

## LIST OF TABLES

Table 4.1-1: GNF2 Fuel Bundle Key Attributes.....	4-4
Table 4.1-2: Analytical Techniques in Core Design .....	4-5
Table 4.2-1: GNF2 Fuel Assembly Material.....	4-23
Table 4.3-1: Cycle Independent Safety Analyses and Operating Envelope Evaluations Potentially Affected by Operating cycle Length .....	4-35
Table 4.4-1: Typical Thermal–Hydraulic Design Characteristics of the Reactor Core .....	4-55
Table 4.6-2: Control Rod Insertion Versus Time.....	4-73

## LIST OF FIGURES

Figure 4.1-1: BWRX-300 RPV and Internals .....	4-6
Figure 4.1-2: GNF2 and GE14 Lattice Array .....	4-7
Figure 4.2-1: Fuel Assembly .....	4-24
Figure 4.2-2: Ultra-HD-Control Rods Distributed Throughout Core.....	4-25
Figure 4.3-1: Core Map and Preliminary Instrumentation Layout.....	4-35
Figure 4.3-2: Core Loading Map – Reference Equilibrium Cycle .....	4-36
Figure 4.3-3: Representative Control Rod Patterns .....	4-37
Figure 4.3-4a: Peak to Average Bundle Power Ratio at BOC (Average Bundle Power – 3.625 MW) .....	4-38
Figure 4.3-4b: Peak to Average Bundle Power Ratio at MOC (Average Bundle Power – 3.625 MW) .....	4-39
Figure 4.3-4c: Peak to Average Bundle Power Ratio at EOC (Average Bundle Power – 3.625 MW) .....	4-40
Figure 4.3-5: Axial Power Shapes .....	4-41
Figure 4.3-6: Reference Equilibrium Cycle Thermal Margin – Minimum Critical Power Ratio .....	4-42
Figure 4.3-7: Reference Equilibrium Cycle Thermal Margin – MLHGR.....	4-43
Figure 4.3-8: Reference Equilibrium Cycle Thermal Margin – MFLPD .....	4-44
Figure 4.3-9: Shutdown Margin vs Cycle Exposure .....	4-45
Figure 4.4-1: BWRX-300 Core Inlet Orifice Type Arrangement .....	4-56
Figure 4.4-2: BWRX-300 Power / Flow Map.....	4-57
Figure 4.4-3: Relative Power for Analyzed Node (hot channel and average channel) .....	4-58
Figure 4.4-4: Relative Void Fraction for Analyzed Node (hot channel and average channel) .....	4-59
Figure 4.6-1: CRD Interfaces Diagram .....	4-74
Figure 4.6-2: Magnet Coupling Fine Motion Control Rod Drive Cross-Section .....	4-75
Figure 4.6-3: Schematic Representation of Magnet Coupling Fine Motion Control Rod Drive .....	4-76
Figure 4.6-4: Representative Continuous Full-in Indicating Device.....	4-77
Figure 4.6-5: Representative Control Rod Separation Detection .....	4-78
Figure 4.6-6: Representative Control Rod to Control Rod Drive Coupling .....	4-79
Figure 4.6-7: Representative FMCRD Electro-Mechanical Brake .....	4-80
Figure 4.6-8: Representative Internal CRD Blowout Support Schematic .....	4-81
Figure 4.6-9: HCU Simplified PID.....	4-82
Figure 4.6-10: CRD Simplified P&ID .....	4-83

Figure 4.6-11: Representative FMCRD Anti-Rotation Devices .....	4-84
---	------



## **4.0 REACTOR**

Chapter 4 describes the components of the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal-hydraulic design including requirements pertaining to safety.

The reactor assembly consists of the Reactor Pressure Vessel (RPV), pressure-containing appurtenances including Control Rod Drive (CRD) housings, in-core instrumentation housings, and reactor internal components (Figure 4.1-1). A summary of the important design and performance characteristics of the reactor and plant is given in Table 1.5-1.

A brief overview of the BWRX-300 reactor is provided in Section 4.1. The reactor materials, internal components, specifications, controls on welding, non-destructive examination, fabrication, and materials are described in Chapter 5, Section 5.2. The reactor internal components are described in Chapter 5, Subsection 5.4.2. The RPV design is described in Chapter 5, Section 5.4.

The fuel assembly and control rods are described in Section 4.2 including the design basis requirements, analytical methods, and evaluation results. Aspects pertaining to the nuclear design of the core, including the reference BWRX-300 design used for safety analysis, are described in Section 4.3. The thermal-hydraulic design basis requirements and important methodologies are described Section 4.4, and thermal-hydraulic stability is discussed in Section 4.8. The core monitoring function is described in Section 4.7. A description of the CRD system and associated requirements is provided in Section 4.6.

The design bases and functional requirements applied in the nuclear design of the fuel, core and reactivity control system comply with CNSC REGDOC-2.5.2 (Reference 4.1-1), Section 8.1.

### **4.1 Summary Description**

Chapter 4 describes the components of the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal-hydraulic design including requirements pertaining to safety.

The reactor assembly consists of the RPV, pressure-containing appurtenances including CRD housings, in-core instrumentation housings, and reactor internal components (Figure 4.1-1). A summary of the important design and performance characteristics of the reactor and plant is given in Table 1.5-1.

#### **4.1.1 Reactor Pressure Vessel**

The BWRX-300 RPV is a vertical, cylindrical pressure vessel fabricated with forged rings or rolled plate welded together with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles, and reactor internals support.

The increased RPV height, relative to forced circulation Boiling Water Reactors (BWRs), is achieved by a “chimney” in the space that extends from the top of the core (top guide) to the entrance to the chimney head and steam separator assembly. The natural circulation flow resulting from the tall RPV results in adequate thermal margins during power operation and off-normal conditions as described in Section 4.3.

The RPV design and description are provided in Chapter 5, Section 5.4.

#### **4.1.2 Reactor Internal Components**

The major reactor internal components are described Chapter 5, Section 5.4, including:

- Core components (control rods and nuclear instrumentation)
- Core support structures (shroud, shroud support, top guide, core plate, control rod guide tubes, control rod drive housings, fuel support castings, and orificed/peripheral fuel supports)
- Chimney
- Chimney head and steam separator assembly
- Steam dryer assembly

Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion-resistant stainless steels or other high alloy material. The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, chimney assembly, steam dryers and in-core instrumentation assemblies are removable when the reactor vessel is opened for refueling or maintenance.

#### **4.1.3 Reactor Core**

The reactor core is comprised of 240 fuel assemblies arranged to form an upright cylinder. Additionally, movable control elements are inserted or withdrawn for reactivity control. The fuel assemblies are comprised of hermetically sealed fuel rods in a square array along with upper & lower tie plates, water rods, fuel rod spacers, fuel channel and connecting components. The fuel assemblies are supported by the reactor internals. The reactor internals also direct the flow of the coolant into the fuel assembly past the fuel rods. The coolant is light water and serves as both the working fluid and moderator. Boiling within the fuel assembly results in a two-phase mixture that exits the top of the fuel assembly that then enters the steam separators after traversing the axial extent of the chimney. The resulting dry steam at the exit of the steam dryers is then sent to the turbine to drive the turbine generator. The RPV and major components are depicted in Figure 4.1-1.

#### **4.1.4 Fuel Assembly and Control Rod Assembly**

A BWRX-300 GNF2 fuel assembly consists of 92 fuel rods and two large central water rods that occupy eight (8) fuel rod locations contained in a 10x10 array (i.e., 100 lattice locations). Fourteen fuel rod locations are occupied by part length fuel rods.

The fuel rod consists of uranium dioxide in the form of cylindrical pellets contained in Zircaloy tubing. The tubing is plugged, sealed, and welded at the ends to encapsulate the fuel. Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown and promote heat transfer.

Key attributes of the GNF2 fuel assembly are summarized in Table 4.1-1.

The control rods are cruciform shaped and reside in the intra-assembly gap. As such, the control rod geometric envelope is separated from that of the fuel assembly. A control rod is associated with four (4) fuel assemblies that, together, constitute the fundamental BWR cell.

The design of the fuel and control rods is described in Section 4.2.

The CRD mechanisms used to withdraw and insert the control rods are of the fine motion type providing fine reactivity control, fast scram and redundant insertion means. The CRD system is described in Section 4.6

#### **4.1.5 Nuclear Instrumentation**

The performance of the core is monitored by fixed neutron detectors within the core. The in-core nuclear instrumentation provides input to automatic reactor core control functions. The BWRX-300 nuclear instrumentation consists of Power Range Neutron Monitoring System (PRNM), Gamma Thermometers (GTs), and Wide Range Neutron Monitors.

The PRNM provides neutron monitoring power signals to the SC1 I&C protection systems. The PRNM also provides signals for post-accident monitoring purposes and, through isolated one-way optical data, links to the core monitoring three-dimensional power distribution program and to the control rod blocking systems.

GTs are in-core devices that convert local gamma flux to an electrical signal that supplies information required to calibrate the Local Power Range Monitors (LPRM) in the PRNM system. They also provide input to the SC2 I&C protection systems for various trip and monitoring functions.

The Wide Range Neutron Monitor system is a redundant pair of industrial computers with a real time operating system that monitors the fixed neutron detectors in the core. The detectors are distributed radially in the core at fixed heights. Each detector is sensitive to neutrons from below criticality to approximately 20% thermal power. The core monitoring function design is provided in Sections 4.7 and Chapter 7, Section 7.3.

#### **4.1.6 Analysis Techniques**

The analytical techniques employed in core design are comprised of the computer codes summarized in Table 4.1-2 and engineering design practices. The computer codes in Table 4.1-2 are further described in Chapter 3, Appendix 3G.

#### **4.1.7 References**

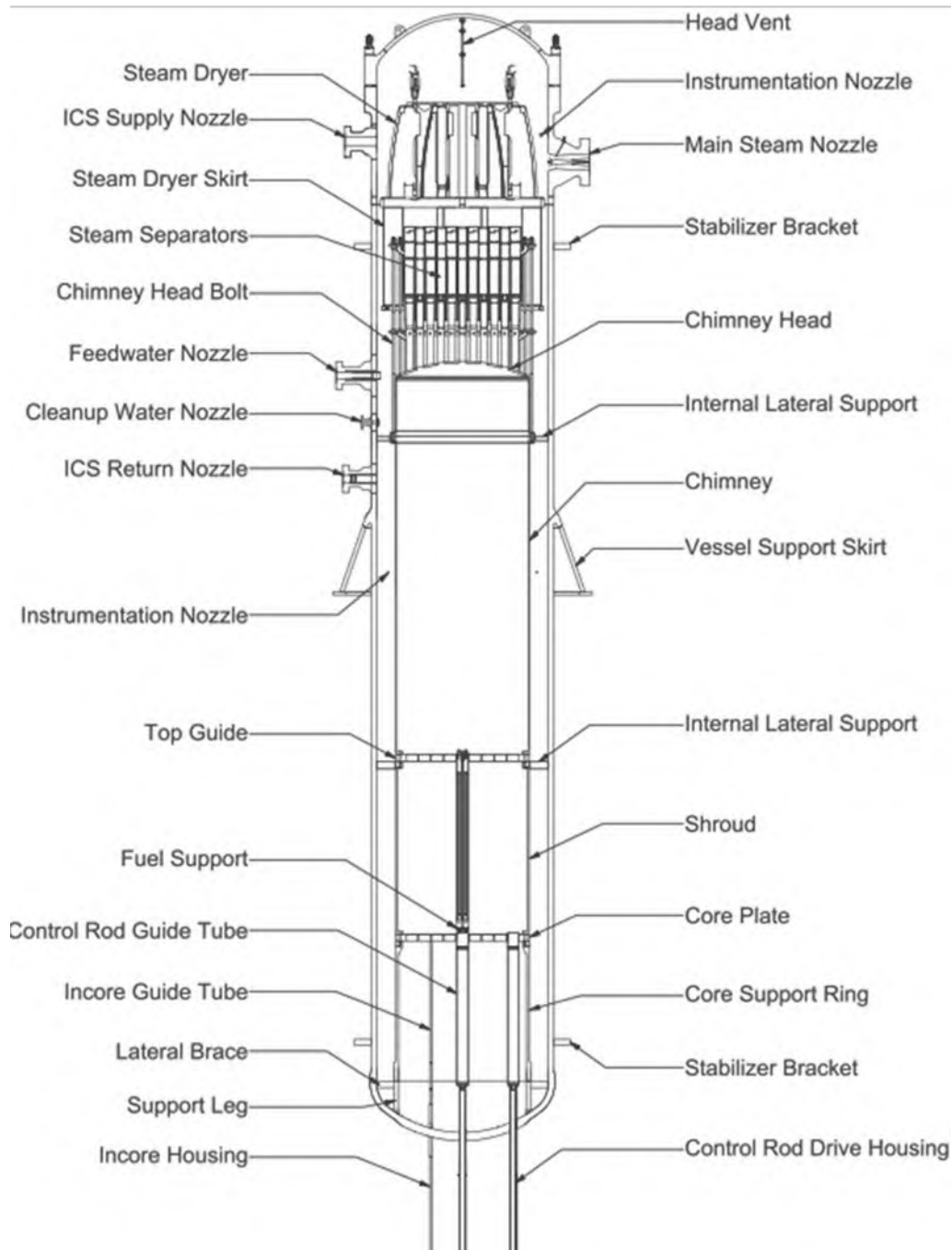
- 4.1-1 CNSC Regulatory Document REGDOG-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

**Table 4.1-1: GNF2 Fuel Bundle Key Attributes**

	<b>GNF2</b>
<b>Fuel Bundle</b>	
Total number of fuel rods	92
Number of full-length rods	78
Number of partial length rods	14 total, Two Lengths
Number of part length rods (long)	8
Number of part length rods (short)	6
Lattice Array	Figure 4.1-2
Representative Assembly weight (kgU)	186
<b>Fuel Rod</b>	
Cladding material	Zr-2 with zirconium inner liner
Assembly active fuel length (mm)	3810
Fuel pellet Material	UO <sub>2</sub>
Fuel Rod Fill Gas	He
<b>Water Rod</b>	
Number of Water Rods	2
Water Rod Material	Zr-2
<b>Spacer</b>	
Spacer Structure	Grid-type with flow diverters
Spacer Material	Alloy X-750
Number of spacers	8

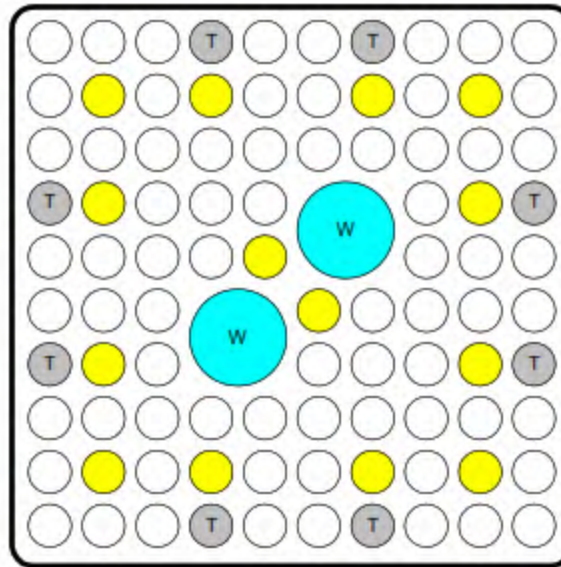
**Table 4.1-2: Analytical Techniques in Core Design**

Analysis	Technique	Computer Code
Fuel rod design	Numerical solutions of 1D steady state and transient heat transfer and finite element mechanical analysis	PRIME03P
Fuel performance characteristics (temperature, internal pressure, and clad stress, etc.)		
Nuclear design Cross-sections and group constants	Lattice physics	TGBLA06
X-Y and X-Y-Z power distribution, reactivity coefficients	Steady state coupled nuclear thermal hydraulics Quasi 2 group diffusion theory	PANAC11
Axial power distributions, control rod worth		
Fuel rod power		
Thermal-hydraulic design steady state	Multi-dimensional, two-fluid model thermal hydraulics 3-D reactor kinetics	TRACG

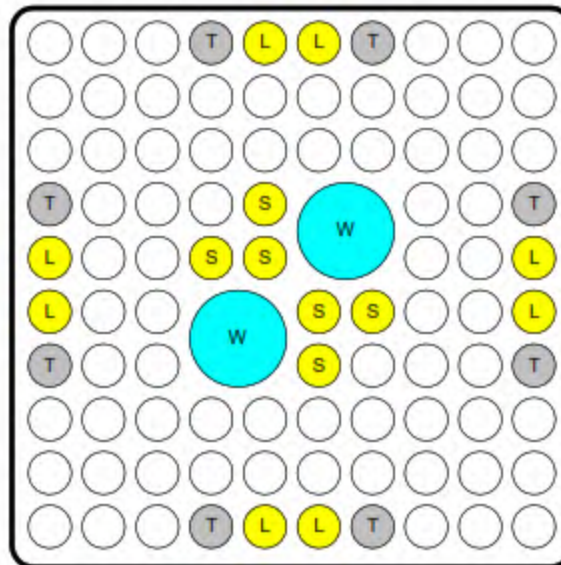


**Figure 4.1-1: BWRX-300 Reactor Pressure Vessel and Internals**

**GE14**



**GNF2**



- water rod
- PLR - 2133.6 mm
- PLR - 2590.8 mm
- PLR - 1371.6 mm
- tie rod

**Figure 4.1-2: GNF2 and GE14 Lattice Array**

## **4.2 Fuel System Design**

The fuel system consists of the fuel assembly and the reactivity control assembly (i.e., the control rods). The fuel assembly is comprised of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle. The fuel bundle is comprised of fuel rods (some of which contain burnable neutron absorbers), upper and lower tie plates, water rods, spacers, springs, and assembly fittings.

The BWRX-300 reactor core is fueled with commercially available BWR fuel currently in production for forced circulation BWRs (e.g., GE BWR/4-6 and Advanced Boiling Water Reactor (ABWR)). The reference fuel design for reactor licensing is GNF2 (Figure 4.2-1). GNF2 is the third evolution of the Global Nuclear Fuel (GNF) 10x10 fuel design and was first deployed in batch quantities in 2007

GNF2 fuel conforms with the requirements stipulated in Licensing Topic Report "General Electric Standard Application for Reactor Fuel" (Reference 4.2-1). GEH BWR fuel designs have been developed to conform to the New Fuel Licensing requirements stipulated in GESTAR as per (Reference 4.2-1) NEDC-33270P. GESTAR is the framework for implementing U.S. regulatory requirements and consistent with international standards as evidenced by the multiple jurisdictions in which GNF fuel has been deployed. Furthermore, the requirements in GESTAR are evaluated in (Reference 4.2-2) which shows that the fuel meets the requirements in the CNSC REGDOC-2.5.2. (Reference 4.1-1).

### **4.2.1 Design Bases**

#### **4.2.1.1 Fuel Assembly Design Bases**

The fuel assembly design bases are established to satisfy the performance and safety criteria presented in CNSC REGDOC-2.5-2 (Reference 4.1.-1).

The fuel assembly is designed to:

- Support self-sustaining fission chain reaction, thereby producing energy in the form of heat
- Maintain its integrity to retain fission products generated in the fuel during normal operation and operational transients or during Anticipated Operational Occurrence AOO conditions

The fuel rod design considers all applicable effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is achieved by designing to prevent excessive fuel temperatures, internal gas pressure due to fission gas releases, cladding stresses, strains, and strain fatigue. The fuel rods are designed so that the conservative design bases of the limiting set of events envelop the lifetime operating conditions of the fuel. For each design basis, the performance of the limiting fuel rod, with appropriate consideration for uncertainties, does not exceed the limits specified by the design basis (as discussed in Section 4.2). The detailed fuel design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre-pressurization level.

The fuel assembly structure integrity is assured by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. The fuel assembly is designed to withstand the following loads:

- Non-operational loads occurring in shipping and handling
- Normal and abnormal loads occurring in startup testing, normal operation, AOOs
- Abnormal loads occurring in infrequent events and accidents



Chapter 4 establishes the fuel design criteria. Specifically, Section 4.2 identifies fuel damage criteria. Section 4.3 establishes fuel criteria for axial offset anomaly. Section 4.4 provides specific thermal-hydraulic criteria.

#### **4.2.1.2 Control Rods Design Bases**

The control rods are designed to control the fission chain reaction. The rods, along with the control rod drive system (Section 4.6), provide stable and automatic control of reactor core power, spatial instabilities, and local power density during normal operation. The control rods also shut down the reactor and maintain the core subcritical.

The control rod design meets the following acceptance criteria:

- Control rod stresses, strains, and cumulative fatigue are evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection
- The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses
- Control rod materials are shown to be compatible with the reactor environment
- Control rod reactivity worth is included in the plant core analyses

The following bases are established for the above acceptance criteria.

#### **Stress, Strain, and Fatigue**

The control rod design is evaluated to assure that it does not fail because of loads due to shipping, handling, normal operation, including the effects of AOOs, Postulated Accidents (PAs) and Design Extension Conditions (DECs). To ensure that the control rods do not fail, these loads must not exceed the ultimate stress and strain limit of the material, structure, or welded connection. Fatigue must not exceed a fatigue usage factor of 1.0.

The loads evaluated include those from normal operational transients (scram and rod maneuvering), pressure differentials, thermal gradients, flow, and system induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time, as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

#### **Control Rod Insertion**

The control rod design is evaluated to assure that it can be inserted during normal operations including the effects of AOOs, PAs and DECs. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiation growth.

#### **Control Rod Material**

The external control rod materials must be capable of withstanding the reactor coolant environment for the life of the control rod. Effects of crud deposition, crevices, stress corrosion and irradiation upon the material must be included in the control rod design and core evaluations. Irradiation effects to be considered include material hardening and absorber depletion and swelling.

## **Reactivity**

The reactivity worth of the control rod design is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance must also be included in the plant core analyses including the effects of normal operations, AOOs, PAs and DEC's.

### **4.2.2 Fuel Assembly and Control Rods Description**

#### **4.2.2.1 Fuel Assembly Description**

The reference GNF2 fuel assembly components are shown on Figure 4.2-1, and consists of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle. The fuel rods and water rods are spaced and supported by upper and lower tie plates and intermediate spacers. The lower tie plate has the function of supporting the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel bundle from one location to another.

The identifying fuel assembly serial number is engraved on the top of the handle with no two assemblies bearing the same serial number. This unique numbering supports the programmatic requirements for nuclear material accountancy required in CNSC REGDOC-2.13.1, Safeguards and Nuclear Material Accountancy, Section 7 (Reference 4.2-12). A projection from one side of the handle ensures proper orientation of the assembly in the core.

The materials used in the GNF2 fuel assemblies are listed in Table 4.2-1. Water chemistry controls used to minimize adverse effects on materials are described in Chapter 9A, Sections 9A.8, 9A.9, and 9A.10 and Chapter 13, Subsection 13.3.2.

#### **4.2.2.2 Fuel Bundle**

The GNF2 fuel bundle is comprised of fuel rods that contain a cladding tube that house the  $\text{UO}_2$  fuel pellets with some  $\text{UO}_2$  pellets containing gadolinia. Each fuel rod is hermetically sealed with welded upper and lower end plugs. The cladding and end plug material is Zircaloy-2 with a zirconium liner for pellet-clad interaction resistance. All fuel rods are inerted with helium before sealing.

##### **4.2.2.2.1 Normal Full-Length Rods**

There are 70 normal full-length rod locations in the GNF2 fuel bundle and reside in holes in the upper and lower tie plates. An expansion spring is installed onto the upper end plug that interacts with the upper tie plate exerts a downward force maintaining the axial position of the fuel rod while accommodating irradiation growth. The active fuel length is 381 cm. The plenum also contains a retention spring that protects the column of fuel pellets during transportation from the manufacturing facility to the reactor site.

##### **4.2.2.2.2 Tie Rods**

Tie rods are normal full-length rods except both the upper and lower end plugs are threaded. The lower end plug is screwed into matching threads that are machined into the lower tie plate. The tie rod upper end plug protrudes through the upper tie plate and is secured with nuts. There are eight (8) tie rods located symmetrically along the bundle periphery.

#### **4.2.2.2.3 *Gadolinia (Burnable Absorber) Rods***

Gadolinia rods are essentially normal full-length rods with normal fuel pellets except sections of the fuel column contain pellets with Gd<sub>2</sub>O<sub>3</sub> homogeneously blended with the UO<sub>2</sub> powder. The resultant pellets that function as a burnable neutron absorber controlling excess reactivity in the fresh fuel. The Gd<sub>2</sub>O<sub>3</sub> concentration is effectively depleted by the end of the first bundle operation cycle and is cycle length dependent.

#### **4.2.2.2.4 *Part Length Rods***

Partial length fuel rods were introduced to reduce the two-phase pressure drop for improved stability (i.e., lower decay ratios as described in Section 4.8) and retained in all subsequent fuel products.

Additionally, partial length fuel rods improve nuclear efficiency by better matching the axial hydrogen-to-fissile uranium ratio (H/U) ratio in BWR fuel with axially varying moderator density. Partial length fuel rods also reduce core reactivity in the cold condition and increase cold shutdown margins.

GNF2 has eight (8) long-partial length fuel rods and six (6) short-partial length fuel rods that have fuel column lengths that are approximately two thirds (2/3) and one third (1/3) of 381 cm (150-inch) active fuel length, respectively. The two lengths are used for fuel efficiency to better match the axial H/U ratio axially as the moderator density decreases. The long partial length fuel rods in GNF2 are located adjacent to the intra-assembly bypass gap (i.e., bundle periphery) and the short partial length fuel rods are located adjacent to the central water rods for improved reactivity margins.

#### **4.2.2.2.5 *Water Rods***

The GNF2 assembly is designed with two large circular water rods that are centrally located and occupy 8 fuel rod lattice positions (i.e., each water rod occupies 4 fuel rod locations).

These holes allow water to be channeled from the bottom end of the bundle to the upper part of the bundle, without boiling, for improved neutron moderation which increase moderation and results in improved reactivity and improves cold shutdown margins. The water rods are hollow Zircaloy tubes with several holes around the circumference near each end allowing coolant flow through. The water rod flow is controlled by the number and diameter of the inlet holes at the lower end. Both water rods are screwed into the lower tie plate, and one is designed to position the eight spacers axially. Tabbed water rods that position the spacers are designed with a D-shaped upper end plug and with spacer positioning tabs. These tabs are welded to the tube exterior above and below each spacer location (see Figure 4.2-1). The tabbed water rod (spacer-positioning) is prevented from rotating by the engagement of its D-shaped upper end plug with a D-shaped hole in the upper tie plate. The non-tabbed water rod has a round shank upper end plug and no spacer-positioning tabs. An expansion spring is located between the water rod shoulder and upper tie plate allowing for differential axial expansion.

The single-phase (i.e., liquid) water that flows through the water rod increases neutron moderation and further thermalizes the neutron energy spectrum, improving nuclear efficiency. Water rods in the fuel assemblies also function to increase moderation for improved reactivity and improved cold shutdown margins.

#### **4.2.2.2.6 Spacers**

The GNF2 fuel design uses 8 spacers with integral springs made of Alloy X-750. The integral spring construction results in significantly fewer parts. The spacer spring force prevents fretting wear on the fuel rods due to fuel rod vibration. Flow diversion devices added to the top of the spacer improves liquid droplet deposition onto the surface of the fuel rods in the two-phase flow region.

The GNF2 bundle design has a non-uniform axial spacing of the 8 spacers optimizing critical power performance. The spacers axial separation in the lower region of the bundle prevents excessive fuel rod bow during operation. The spacers are positioned closer to each other in the upper region of the bundle for increased liquid droplet deposition in the annular flow regime. The GNF2 spacer exhibits low hydraulic resistance (i.e., pressure drop). Alloy X-750 is resistant to hydrogen pickup and maintains strength and ductility through the operating lifetime and while in storage.

#### **4.2.2.2.7 Upper Tie Plate**

The type-304 stainless steel upper tie plate supports the weight of the fuel assembly and positions the rod ends laterally during operation and handling. The upper tie plate has an internally threaded corner post fabricated to accept the channel fastener bolt. The upper tie plate is synergistically designed with two large central water rods and the partial length fuel rods maximizing flow area and minimizing two-phase pressure-drop.

#### **4.2.2.2.8 Debris Filter Lower Tie Plate**

The type-304 stainless steel lower tie plate, in conjunction with the upper tie plate, supports the weight of the fuel assembly and positions the rod ends laterally during operation and handling. Figure 2.7 of NEDC-33270, GNF2 Advantage Generic Compliance with NEDE-24011-PA (GESTAR II), (Reference 4.2-1) is an assembly consisting of a machined casting, non-removable debris filter cartridge, and a welded cover plate. The debris filter component is specifically designed to maximize filtration efficacy and the hydraulic resistance (i.e., pressure drop) is explicitly modeled in core performance and safety analyses.

#### **4.2.2.2.9 Channel and Channel Fastener**

The BWR fuel channel (Figure 4.2-1) performs the following functions:

1. Forms the fuel bundle coolant flow path outer periphery
2. Provides a surface for control rod guidance in the reactor core
3. Provides structural lateral stiffness to the fuel bundle
4. Controls, in conjunction with the lower tie plate, coolant bypass flow at the channel/lower tie plate interface

The channel is open at the bottom and makes a sliding seal fit over the lower tie plate. At the top of the channel, two opposite corners have welded clips. These clips support the weight of the channel on the upper tie plate posts. One of the clips has a hole for attaching the channel fastener to the bundle. The channel design incorporates a uniform thickness bottom end. The remainder of the channel has corners, sidewalls and sidewall grooves at the control blade roller, and symmetric locations providing sufficient strength in the regions of highest stress while minimizing material that absorbs neutrons.

The overall stiffness of the fuel assembly is almost entirely a function of the channel stiffness. The GNF2 channel stiffness and the fuel assembly stiffness is essentially the same as previous fuel assemblies. The NSF (GNF proprietary zirconium-based alloy channel material) that is standard for GNF2 was introduced in reload quantities in 2016 following U.S. Nuclear Regulatory Commission (USNRC) approval. The design mitigates excessive channel deformation in-service and minimizes channel/control blade interference. The technical basis for replacing Zircaloy-2 and Zircaloy-4 with the NSF channel material is provided in NEDE-33798 (Reference 4.2-3).

The GNF2 channel fastener assembly consists of a stainless-steel casting (or fastener guard), a one-piece Alloy X-750 leaf spring with two active leaves at right-angles, a spring lock washer, and a fastener bolt attaching the fuel channel to the upper tie plate.

The channel fastener casting (or guard) fits over the top of the channel and bolts through the channel clip into the upper tie plate. The fastener guard serves as a reaction support for the leaf springs, provides a captive housing and lead-in for the fastener spring, and protects the springs from being overstressed. The fastener bolt attaches the channel to the bundle and remains captive in the casting, even if the fastener bolt were to fail.

#### **4.2.2.3 Control Rods**

The control rods (Figure 4.2-2) perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of control rods. These rods are positioned to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all control rods be available for either reactor “scram” (i.e., prompt shutdown) or reactivity control. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, electro-hydraulically actuated drive mechanisms that allow either electric motor controlled axial positioning for reactivity regulation or hydraulic scram insertion. The design of the rod-to-drive connection permits each control rod to be coupled or uncoupled from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and remain operable for tests with the reactor vessel open.

The core reactivity control requirements are met by use of the combined effects of the movable control rods, supplementary burnable neutron absorber (i.e., Gadolinia Rods in the fuel bundle described in Subsection 4.2.2.2.3), and the reactor coolant natural flow.

The control rod main structure consists of a top handle, an absorber section, and a bottom connector assembled into a cruciform shape. The top handle contains a grapple opening for handling. The absorber section is an array of stainless-steel tubes filled with boron carbide powder capsules or a combination of boron carbide powder capsules and hafnium rods. The connector is positioned on the bottom of the control rod for coupling to the control rod drive. While being inserted into the core, the control rod is restricted to the cruciform envelope created by the fuel bundles. Handle pads guide the control rod along the channels and connector rollers guide the control rod within the guide tube as the control rod is inserted and withdrawn from the core. Configuration of the control rod is shown in Figure 4.2-3.

The BWRX-300 employs the Ultra-HD control rod, which is based on the Ultra-HD control rod designed for the BWR/2 through BWR/6. That design has been approved by the USNRC and applied to operating plants. The GEH Ultra-HD control rod is a derivative of prior designs incorporating the full-length hafnium rods in outer edge, high depletion tube locations. A detailed description is given in (Reference 4.2-4), NEDE-33284 Supplement 1P-A, "Marathon-Ultra Control Rod Assembly."

#### **4.2.3 Fuel Design Evaluation**

The GNF2 fuel design is the result of over 50 years of BWR fuel design, fabrication, and operational experience. The BWR fuel design process, comprised of engineering methods, analyses and test, is mature and generically applicable to the BWRX-300. The methods applied in the design of BWR fuel are documented in various Licencing Topical Reports that were submitted to the USNRC as needed. In addition, GNF developed a fuel licensing framework with the USNRC, and other regulatory authorities, called GESTAR, NEDE-24011-P-A-31 (Reference 4.4-8). The GESTAR licensing framework defines the generic requirements and approved methods for designing BWR fuel.

Most of the fuel rod thermal mechanical design analyses are performed using the PRIME fuel rod thermal-mechanical methodology. The method, qualification and application of the PRIME methodology are described in NEDC-33256P-A, NEDC-33457P-A, NEDC-33458P-A and NEDC-33840P-A (References 4.2-5, through 4.2-8). The PRIME methodology is approved for application to AOO transients (Reference 4.2-8, NEDC-33840P-A). This methodology is applicable to the analysis of the fuel rod response for all transient events and has been qualified for performing fuel analysis for BWRX using US NQA-1 standard, which is equivalent to Canadian standard CSA N286.7. PRIME analyses are performed for the following conditions:

1. For each analysis, fuel rod input parameters are based on either the most unfavorable manufacturing tolerances (i.e., 'worst tolerance' analyses) or statistical distributions of the input values. Calculations are performed providing either a 'worst tolerance' or statistically bounding tolerance limit for the resulting output parameter(s)
2. Operating conditions are postulated that cover the conditions anticipated during normal operating conditions and AOOs

Fuel rod design evaluations establish an upper bound power history envelope for the different fuel rod types. These power histories are used for fuel rod thermal mechanical design analyses evaluating the fuel rod design features and demonstrating conformance to the design criteria. These power histories are applied as a design constraint in the development of any BWR core design, including the reference BWRX-300 core described in 4.3.

The PRIME analysis assumes that during the fuel rod operating lifetime, the fuel rod (axial) node with the highest power operates on the limiting power-exposure envelope.

##### **4.2.3.1 Worst Tolerance Analyses**

The analyses apply worst tolerance assumptions to the cladding circumferential strain during an AOO. In this case, the important PRIME inputs are all biased to the fabrication tolerance extreme in the direction that produces the most limiting circumferential strain. The biases are discussed in detail in NEDC-33258P-A (Reference 4.2-7).

##### **4.2.3.2 Statistical Analyses**

For PRIME analyses are performed using standard error propagation statistical methods. The statistical analysis procedure is presented in NEDC-33258P-A (Reference 4.2-7).

#### **4.2.3.3 Fuel Lift and Seismic and Dynamic Load Analysis**

The fuel lift and seismic and dynamic load analyses are completed prior to fuel release for transportation. The fuel lift seismic and dynamic load analysis evaluates whether these loads are sufficient to unseat the fuel assembly from the lower tie plate, and if so, the maximum vertical lift distance and maximum fuel dynamic acceleration resulting from reseating. The acceptance criteria for this analysis are:

1. Vertical fuel lift is less than the engagement between the fuel support and the lower tie plate
2. Peak horizontal and vertical accelerations are less than the fuel demonstrated acceleration capabilities

#### **4.2.3.4 Cladding Strain**

The cladding strain analysis is performed using the PRIME code and the worst tolerance methodology discussed previously. For each fuel rod type, as described in Subsection 4.2.2.2, the cladding strain is calculated at various exposure points where transient overpower is assumed relative to the limiting power history. The magnitude of the overpower event is increased until the cladding strain approaches, but does not exceed, the limits described in NEDC-33258P-A (Reference 4.2-7) at the most limiting exposure to establish the mechanical overpower. Mechanical overpowers may also be defined as a function of exposure, ensuring that cladding strain does not exceed the limits described in NEDC-33258P-A (Reference 4.2-7) at any exposure point.

#### **4.2.3.5 Fuel Rod Internal Pressure**

The fuel rod internal pressure analysis is performed using the PRIME code and the statistical methodology discussed previously. The fuel rod internal pressure nominal value and standard deviation are determined at various fuel rod exposure points. At each of these exposure points, the nominal values and standard deviation of the fuel rod internal pressure that would cause the cladding to creep outward at a rate equal to the fuel pellet irradiation swelling rate (the critical pressure) are calculated. A design ratio is defined from the rod internal pressure and the critical pressure with their standard deviations. This design ratio limit of no more than 1.0 at a 95% confidence level, ensures that the fuel rod cladding does not creep out at a rate greater than the fuel pellet irradiation swelling rate.

#### **4.2.3.6 Fuel Pellet Temperature**

The fuel pellet temperature analysis is performed statistically using the PRIME code. For each fuel rod type the fuel pellet centerline temperature is statistically calculated at various exposure points assuming an overpower relative to the limiting power history. The lower 95% value for margin to fuel melting is calculated based on these statistical distributions. The overpower event magnitude is further increased until the lower 95% melt margin approaches zero but remains positive at the most limiting exposure point. The result from this analysis establishes the thermal overpower discussed below. Thermal overpowers may also be defined as a function of exposure, ensuring that lower 95% melt margin remains positive at all exposure points.

#### **4.2.3.7 Cladding Fatigue Analysis**

The cladding fatigue analysis is performed statistically using the PRIME code. Variations in power, coolant pressure and coolant temperature are superimposed on the limiting power history for calculating the cladding fatigue.

Cladding strain cycles are analyzed in NEDC-33270P (Reference 4.2-1) using the fuel duty cycles shown in NEDC-33258P-A (Reference 4.2-7). The duty cycles represent conservative assumptions regarding power changes anticipated during normal reactor operation and AOOs, planned surveillance testing, normal control blade maneuvers, shutdowns and load following. Fatigue analyses explicitly account for daily load following using the BWRX-300 plant duty cycles specification, with 347 cycles per year. The analyses of GNF2 fuel show that the cladding fatigue capability accommodates the expected fatigue duty for the projected life and operation of the fuel with sufficient margin (Reference 4.2-11, NEDC-33941P).

#### **4.2.3.8 Cladding Creep Collapse**

Cladding creep collapse occurs when excessive in-reactor fuel pellet densification causes fuel column axial gaps. This phenomenon has been observed in some pressurized water reactor fuel rods. The cladding creep collapse analysis consists of a detailed finite element mechanics analysis of the cladding. This evaluation is described in NEDC-33139P-A (Reference 4.2-9).

#### **4.2.3.9 Fuel Rod Stress Analysis**

The fuel rod stress analysis is performed using the Monte Carlo statistical methodology and addresses local fuel rod stress concerns, such as the stresses at spacer contact points, that are not directly addressed by the PRIME code. Results from PRIME analyses are used to generate inputs for the stress analysis. The cladding stress analysis is described in NEDC-33270-P (Reference 4.2-1).

#### **4.2.3.10 Thermal and Mechanical Overpowers**

Analyses are performed to establish the values of the maximum overpower magnitudes that do not result in violation of the cladding circumferential strain criterion or the incipient fuel centerline melting criterion. As part of the core design and AOO analysis, the calculated core mechanical and thermal overpowers defined in NEDC-33258P-A (Reference 4.2-7) and NEDC-33840P-A (Reference 4.2-8) are compared with the thermal overpower and mechanical overpower criteria and confirm conformance to these criteria. PRIME transient analyses performed per NEDC-33840-P-A (Reference 4.2-8) may also use worst-tolerance and statistical assumptions, consistent with those used in determination of thermal overpower and mechanical overpower criteria. This method confirms compliance by comparing worst-tolerance strain directly to strain limits and lower 95% melt margin criterion.

#### **4.2.3.11 Fretting Wear**

Testing assures that the mechanical design features, particularly those related to spacers and tie plates, do not result in significant vibration and consequent fretting wear, particularly at spacer–fuel rod contact points. The GNF2 fuel design vibration response has been compared to a reference design that has demonstrated satisfactory performance through discharge exposure.

#### **4.2.3.12 Water Rods**

Analyses are performed to determine component stresses at the bounding load conditions and compared to applicable criteria, such as yield and ultimate stresses. The load conditions consider shipping and handling loads, seismically induced bending moment, and the pressure differential across the water rod. The design is also evaluated using finite element analysis to determine the critical buckling load to ensure axial loads resulting from differential growth of water rods and other fuel assembly components are adequate.

#### **4.2.3.13 Tie Plates**

Tie plate adequacy is demonstrated by detailed finite element analysis and mechanical testing for bounding fuel handling and seismic load conditions.



#### **4.2.3.14 Spacers**

Tests are performed to demonstrate that the spacer design can withstand design basis loading without any significant deformation. The bounding load condition is seismic loading. Tests are conducted to demonstrate spacer fatigue capability and compliance with load limits and to demonstrate that a coolable geometry is maintained by showing minimal deformation at the combined load condition. Fretting wear is addressed by performing flow induced vibration tests and evaluating the results relative to spacer designs that have demonstrated acceptable performance. The results of flow induced vibration testing for GNF2 are summarized in NEDC-33270P (Reference 4.2-1).

#### **4.2.3.15 Channel**

Channel adequacy relative to applicable design criteria is confirmed by performing the following evaluations:

- Calculating elastic stress and deflection caused by channel wall pressure difference
- Calculating thermal stresses due to the various temperature gradients that the channel is subjected to during normal operation and handling
- Calculating fatigue and stress rupture considering the combined effect of pressure temperature cycling and hold time
- Calculating elastic-plastic and creep of channel wall permanent deflection
- Calculating channel stress due to control rod contact
- Analyzing channel/lower tie plate differential thermal expansion

The GNF2 Fuel Assembly Mechanical Design Report for BWRX-300 (Reference 4.2-10) provides additional discussion addressing the design features that preclude excessive channel bowing from preventing control blade insertion.

#### **4.2.3.16 Conclusion**

The BWRX-300 reactor is designed to accept conventional BWR fuel and GNF2 has been established as the reference fuel design. Fuel mechanical and fuel rod thermal mechanical evaluations, summarized in NEDC-33270P (Reference 4.2-1), demonstrate compliance with regulatory requirements and are applicable to GNF2 operation in the BWRX-300. Detailed technical information pertaining to fuel assembly mechanical (NEDC-33940P) and fuel rod thermal-mechanical performance (NEDC-33941P) is provided in (References 4.2-10 and 4.2-11).

#### **4.2.4 Control Rods Design Evaluation**

The control rod design is evaluated against the acceptance criteria and bases described in Subsection 4.2.1.2. Design compliance with these criteria constitutes the basis for acceptance and approval of the design. The control rods for the BWRX-300 are based on the design used in the operational BWR fleet. This well understood fleet operational data is utilized in the methods to design, evaluate, and analyze the control rods.

#### **4.2.4.1 Scram**

The dynamic loads on the control rods are bounded by the Fine Motion Control Rod Drive (FMCRD) imposed loads (scram loads) in the vertical direction. The BWRX-300 inoperative buffer loads are the highest vertical loads experienced by the control rod due to the high terminal velocity. The control rod is evaluated using a dynamic analysis in (Reference 4.2-4). The resultant loads are evaluated using the material properties and geometry for the area subject to the load. The effective stress is determined using distortion energy theory. The limit is the material ultimate stress or strain.

#### **4.2.4.2 Seismic**

Fuel channel deflections which result from seismic and Loss-of-Coolant Accident events impose lateral loads on the control rods (Reference 4.2-4). The BWR/2 through BWR/6, ABWR and BWRX-300 have similar channel lengths and deflections.

The seismic analysis is performed by evaluating the strain in the Ultra-HD control rod absorber section when deflected. During a seismic event, it is assumed the seismic deflections could be added to any preexisting channel bow. The absorber section strain has been analyzed for channel deflections due to seismic and channel bow deflections when deflected by a bounded value and found to be acceptable, (Reference 4.2-4).

Testing was performed on the ABWR Marathon control rod to confirm seismic scram capability. The ABWR Marathon control rod was tested at fuel channel oscillation amplitudes representing SSE conditions. The scram times were found to be acceptable, and the control rod was not damaged. The BWRX-300 channels will experience reduced creep bulge as the channel wall pressure differential (i.e., the pressure differential across the channel wall) is low due to natural circulation flow. Since the Ultra-HD geometry and bending stiffness are nearly identical to the tested Marathon, the ABWR Marathon control rod seismic scram capability testing is conservative in relation to the BWRX-300 conditions.

#### **4.2.4.3 Stuck Rod**

Compression caused by a stuck rod at the time of scram is controlled by the FMCRD. Assuming the FMCRD exerts the same compression loads, the control rod does not buckle if the entire load is applied to one wing, (Reference 4.2-4).

#### **4.2.4.4 Absorber Irradiated Related Loads**

The absorber containment licenced in NEDE-33284 (Reference 4.2-4) is applicable to the BWRX-300 Ultra HD control rod. The same methodology is used for the BWRX-300 Ultra HD control rod in (Reference 4.2-4). The absorber tube and B4C capsule design accommodates irradiation-induced swelling of boron carbide, such that a clearance exists between the inner capsule and outer absorber tube up to the maximum local 10B depletion. The analysis conservatively assumes worst-case dimensions plus a bounding boron carbide irradiation swelling rate. As such, all stresses on the outer absorber tube associated with boron carbide swelling are eliminated, which drastically reduces the likelihood of Irradiation-Assisted Stress Corrosion Cracking relative to previous BWR control rod designs. Absorber tube stresses due to helium gas generation and moisture vapor heating are considered. The resulting stresses due to helium pressure are small compared to tube material strengths. Therefore, the Ultra HD absorber containment design is adequate for the nuclear design life of the control rod.

#### **4.2.4.5 Load Combinations and Fatigue**

The BWRX-300 Ultra-HD control rod is designed to withstand load combinations including AOOs and fatigue loads associated with those combinations. Absorber tube loads are evaluated during a scram, in a cell with severe channel bow near end of control rod life when absorber burn-up helium gas generation is highest. Absorber tube loads are evaluated during a seismic event near the end of control rod life when absorber burn up helium gas generation is highest. Absorber section to connector welds and absorber section to handle loads are evaluated during a scram when the absorber helium gas build up is highest. Per NEDE-33284 (Reference 4.2-4), the BWRX-300 Ultra-HD control rod does not exceed the ultimate stress or strain limit of the material. Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of the cyclic loadings in (Reference 4.2-4). The fatigue usage is evaluated against a limit of 1.0.

#### **4.2.4.6 Handling Loads**

The BWRX-300 Ultra-HD control rod is designed to accommodate three times the weight of the control rod, (Reference 4.2-4).

#### **4.2.4.7 Hydraulics**

Inspection experience over 60 years has shown the control rod is not damaged by the vibrations or cavitations set up by coolant velocities and velocity distributions in the bypass region between fuel channels.

#### **4.2.4.8 Materials**

The absorber tubes are made from high purity 304, stainless steel. These tubes are loaded with either, 1) capsules containing B<sub>4</sub>C, or 2) Hafnium rods. Further description is provided in NEDE-33284 (Reference 4.2-4).

Materials selected for use in the Ultra-HD control rod components are chosen to minimize the component end of life radioactivity to reduce personnel exposure during handling on-site, and for final offsite shipping and disposal. All Ultra-HD control rod materials are low weight percent cobalt. The average niobium content for the handle and absorber section, less boron carbide and hafnium has a low weight percent.

Materials with low activation characteristics are specified for use in the control rod. Depleted control rods that are discharged are categorized as high-level radiological waste.

#### **4.2.4.9 Nuclear Performance**

The nuclear lifetime of the initial BWRX-300 control rod type is established as a 10 percent reduction in reactivity worth ( $\Delta k/k$ ) relative to the initial reactivity worth in any quarter axial segment, (Reference 4.2-4).

#### **4.2.4.10 Mechanical Compatibility**

Similar to the control rods supplied for the ABWR and BWR/2 through BWR/6, the BWR X-300 Ultra-HD control rod is designed to be compatible with core and reactor internal interfaces.

The Ultra-HD control rod is designed to be compatible with the Control Rod Guide Tube cylindrical boundary, to provide a seat with the guide tube base during FMCRD removal, to provide lower guide rollers for smooth transitions, and to have clearance with the orificed fuel support for insertion and withdrawal from the core.

The control rod coupling socket provides a compatible interface with the FMCRD. The coupling engages the FMCRD by rotating. With the FMCRD, Control Rod Drive Housing (CRDH), and Control Rod Guide Tube positively assembled, any orientation of the cruciform control rod between the fuel assemblies is a coupled position, and rotation to an uncoupled position is not possible during reactor operation. The four lobes of the FMCRD coupling spud are in line with the four wings of the control rod in the coupled position.

The control rod is designed to permit coupling and uncoupling of the control rod drive from below the vessel for FMCRD servicing without necessitating the removal of the reactor vessel head. The control rod is also designed to allow uncoupling and coupling from above the vessel using control rod handling tools.

The control rod is positively coupled to the FMCRD and is designed to remain coupled during all scrams and loading conditions, including inoperative buffer scram loads. The control rod withstands the loads induced by the FMCRD without exceeding the structural design criteria.

The control rod is dimensionally compatible with the fuel assemblies (unirradiated and irradiated). The control rod is guided, rotationally restrained, and laterally supported by the adjacent fuel assemblies. The control rod is designed and constructed to establish and maintain the alignment of the control rod drive line (Control Rod Drive Mechanisms, Control Rod Guide Tube, and the fuel assemblies) so that control rod insertion and withdrawal is predictable. The top of the active absorber of a fully withdrawn control rod is below the Bottom of the Active Fuel. Absorber gap requirements are placed on the control rod in the operating condition to be compatible with the core nuclear design requirements.

#### **4.2.5 Inspection and Testing**

##### **4.2.5.1 Quality Assurance Program and Quality Control**

The Quality Assurance Program applicable to the fabrication of fuel and control rod assemblies for the BWRX-300 is summarized in Chapter 17.

The program provides for control over activities affecting product quality, starting with design and development, continuing through procurement, materials handling, fabrication, testing, and inspection, storage, and transportation. The program also provides for training of personnel and for the auditing of activities affecting product quality through a formal auditing plan.

The mechanical drawings and product, process, and material specifications identify the inspections performed.

The following subsections describe the general quality control inspections.

##### **4.2.5.2 Fuel System Components and Parts**

The characteristics inspected depend on the component parts. The quality control includes dimensional and visual examinations, review of supplier documentation, material certification, and nondestructive examination. Once supplier processes have been qualified, changes are subjected to GNF review and approval. Suppliers are periodically audited to assure that the process and Quality programs at the manufacturing facilities meet the requirements for GE GNF's Quality program.

The material used in the BWRX-300 core is ultimately accepted and release by Quality Control.

#### **4.2.5.2.1 Pellet**

Pellet Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards. Density is determined using qualified equipment with known verification and calibration standards based. Chemical analyses are performed on a specified sample basis throughout pellet production.

#### **4.2.5.2.2 Fuel Rod Inspection**

Fuel rod inspections consists of the following nondestructive examination techniques and methods, as applicable:

- Each rod is leak tested at the assembly level using a calibrated mass spectrometer, with helium being the detectable gas
- Fuel rod welds are inspected by ultrasonic test in accordance with a qualified technique and GE specifications
- Fuel rods are visually inspected for scratches, dents, and rod bow
- Dimension control on newly fabricated fuel rods is maintained by limiting the allowed variation on the dimensions of the supplied or fabricated components
- Reworked rods are dimensionally inspected prior to final release to verify acceptable length
- Fuel rods are inspected by scanning to verify proper enrichment, burnable absorber poison concentration, and zone length. Scanning also checks for the presence of misplaced pellets or gaps in the fuel column
- Traceability of rods and associated rod components is established by quality control

#### **4.2.5.2.3 Assemblies**

Components for each bundle are visually inspected prior to use in manufacturing.

Bundles are in-process verified during assembly and then final inspected prior to packing into the fresh fuel shipping container.

Shipping containers are licenced and refurbished prior to each use to assure compliance to the Safety Analysis Report.

#### **4.2.5.2.4 Other Inspections**

As part of the routine inspection operation, the following inspections are performed:

1. Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on serialized tools. Complete records are kept of calibration and conditions of tools.
2. Audits are performed of inspection activities and records to confirm that prescribed methods are followed and that records are correct and properly maintained.
3. Surveillance inspection, where appropriate, and audits of outside contractors are performed to confirm conformance with specified requirements.

#### **4.2.5.2.5 Process Control**

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are implemented.

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and unique barcode numbers. Isotopic content is confirmed by analysis.

Powder mixing (enrichment blending or homogenization) is performed to a system generated blend plan. The production control system verifies that only containers in the active blend plan are moved to the process station. The production control system captures the actual weight of each addition to the blend and will abort the blend plan if the weight is out of tolerance. Personnel are assigned roles based upon their work area after qualification/certification; unqualified personnel cannot perform a task as the production software system will prevent the transaction.

Finished pellets are placed on trays identified with a unique barcode identifier that maintains traceability in the production control system. Samples from each pellet lot are tested for isotopic content and impurity levels based upon a sampling plan prior to release of the pellets to rod loading.

Loading of pellets into the cladding is controlled by the production control system which receives the nuclear design from engineering. Each rod is verified to have the correct mass, enrichment, burnable absorber content, and zone length of fuel when the pellets are loaded into the cladding. Traceability is maintained to the pellet trays used to load each zone. Each cladding tube is barcoded, and this barcode is used by the production control system for traceability. An alpha-numeric serial number is placed on the lower plug; this serial number is linked to the barcode identification at the lower end plug ("first") weld process. The upper end plug is inserted and then welded (in an inert gas atmosphere) to seal the tube; during the upper end plug welding the rod is pressurized with Helium as specified by the fuel rod design.

#### **4.2.5.3 Control Rod and Components**

All raw material and purchased component parts are fully inspected and accepted for control rod manufacture. Inspection includes dimensional and visual examinations, review of supplier documentation, material certification, and nondestructive examination. Supplier processes are qualified and cannot be changed without GEH review and approval. Suppliers are periodically audited to assure that the processes and quality programs at the manufacturing facilities meet the requirements for GEH's quality program.

All material used in the BWRX-300 control rods is ultimately accepted and released by GEH supplier quality control.

##### **4.2.5.3.1 Control Rod Absorber Materials**

BWRX-300 control rods use both boron carbide powder and solid hafnium rods as neutron absorbers. Prior to use, boron carbide powder is inspected for chemical content, impurities, water content, isotopic content, and grain size. Several grain sizes are then mixed and vibratory compacted inside stainless steel capsules to achieve the specification-required average density. Similarly, hafnium rods are inspected for chemical content, impurities, and dimensions prior to use. An independent quality overcheck is performed to ensure the correct loading of all absorber materials into each control rod assembly.

#### **4.2.5.3.2 Absorber Tube Inspection**

The manufacture of drawn absorber tubes used in control rod assemblies is controlled to ensure defect-free tubes. Supplier raw material production, welding, drawing and heat treatment process, as well contacting material, and inspection processes, are controlled and approved by GEH. Absorber tubes are inspected both visually and using ultrasonic inspection. Traceability of all absorber tubes is maintained by raw material and tube manufacturing lot.

#### **4.2.5.3.3 Control Rod Assemblies**

The laser weld process used to join absorber tubes into the cruciform control rod assembly is tightly controlled. Periodic weld samples and process tracking are performed to ensure specification weld penetration requirements are met, and to ensure weld quality. Both visual and eddy current inspection is used to ensure continuous welds, with no holes or gaps in finished control rod assemblies. A helium leak check is used to further ensure the integrity of finished control rods.

#### **4.2.6 References**

- 4.2-1 NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-2 NEDC-33977P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Design, Qualification, and BWR Fuel Licensing," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-3 NEDE-33798P, "NEDE-33798P-A Revision 1: Application of NSF to GNF Fuel Channel Designs," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-4 NEDE-33284\_Suppl\_1\_P-A\_Rev\_1, "Marathon-Ultra Control Rod Assembly, Global Nuclear Fuels, Fuel Bundle Designs," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-5 NEDC-33256, "NEDC-33256P-A Revision 2.pdf," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-6 NEDC-33257, "NEDC-33257P-A Revision 2.pdf," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-7 NEDC-33258P-A, "NEDC-33258P-A Revision 2.pdf," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-8 NEDC-33840P, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-9 NEDC-33139PA, "Cladding Creep Collapse," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-10 NEDC-33940P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Mechanical Design Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-11 NEDC-33941P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Thermal-Mechanical Design Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-12 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 4.2-1: GNF2 Fuel Assembly Material**

Component	Material (Reference 4.2-9, NEDC-33139P-A)
Fuel Assembly	
Fuel bundle spacers	Alloy X-750
Water Rods	Recrystallized Zircaloy-2
Tie Plates	Austenitic Stainless Steel
Fuel Rod	
Cladding	Recrystallized Zircaloy-2
Barrier	Recrystallized Zirconium
Pellet	UO <sub>2</sub> and UO <sub>2</sub> - Gd <sub>2</sub> O <sub>3</sub>
Channel and Channel Fastener	
Channel	NSF (GNF Proprietary Zirconium-Based Alloy)
Guard Material	Austenitic Stainless Steel
Spring, cap screw and lock-washer material	Alloy X-750
	XM-19 SST



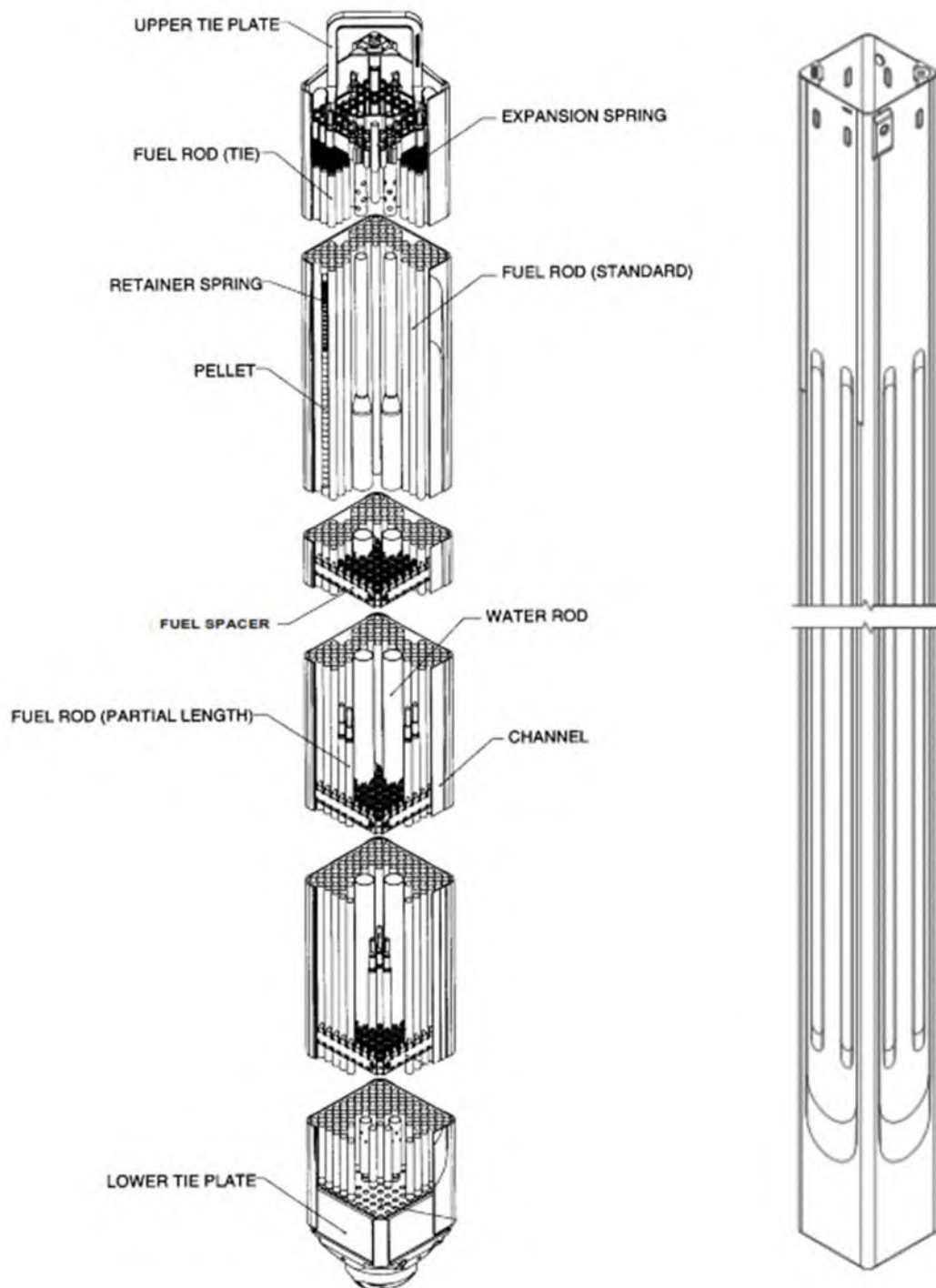
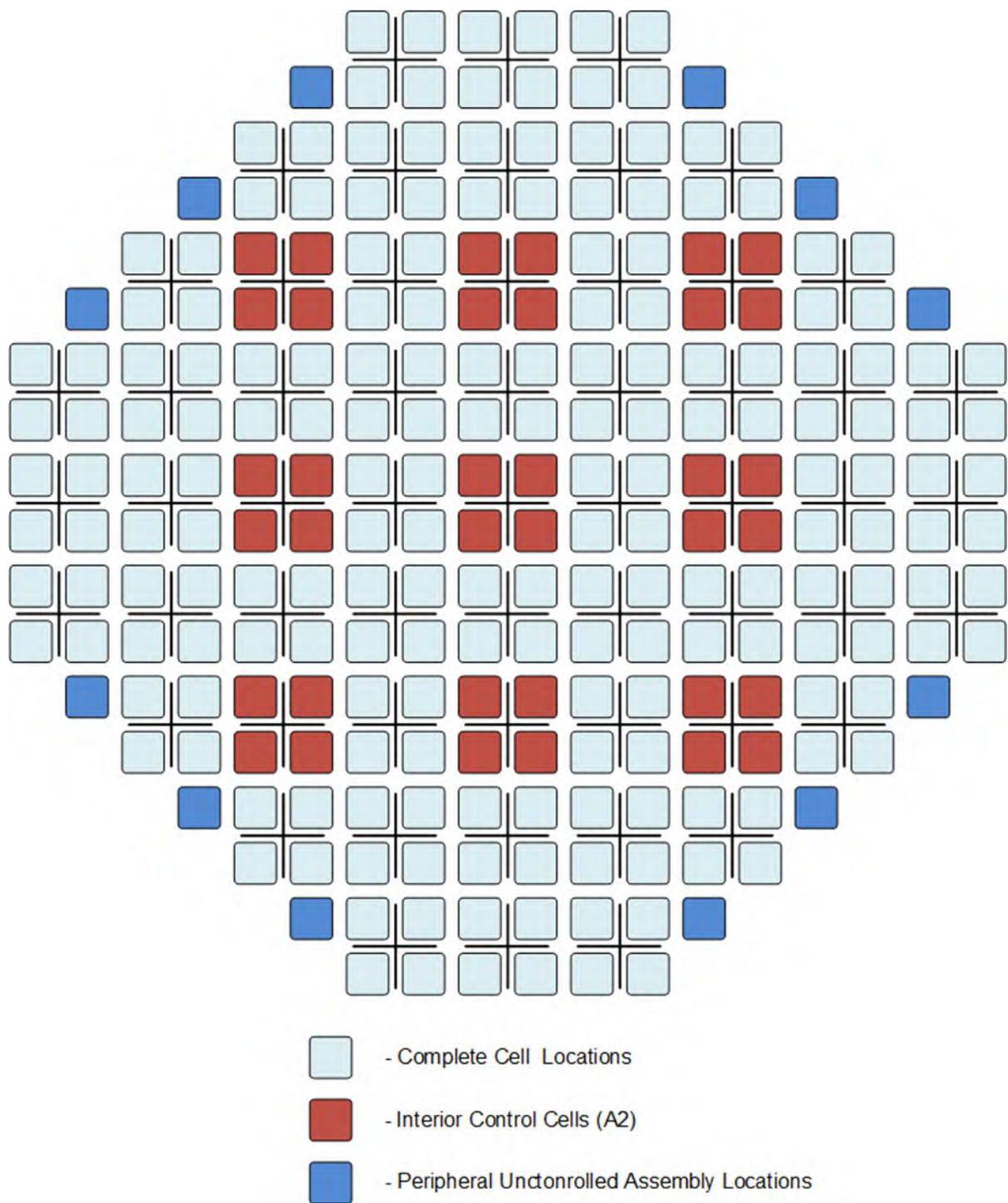
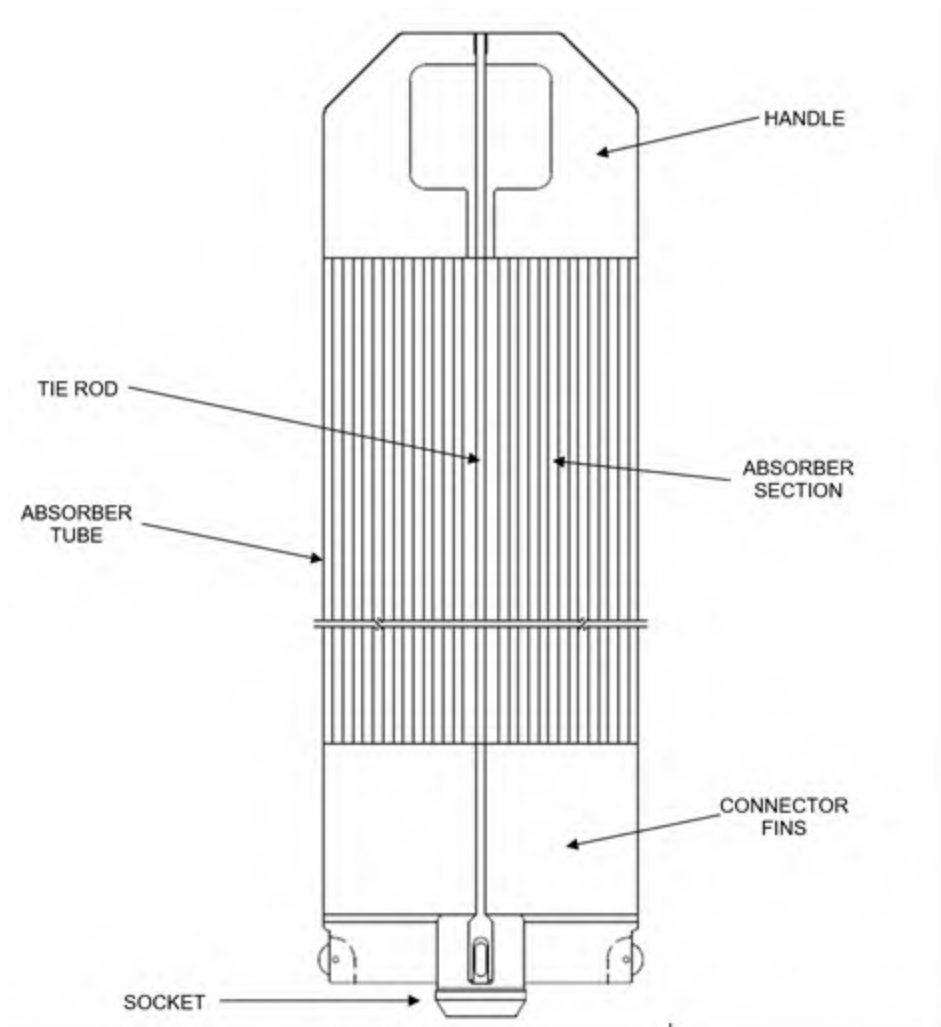


Figure 4.2-1: Fuel Assembly



**Figure 4.2-2: Ultra-HD-Control Rods Distributed Throughout Core**



**Figure 4.2-3 Ultra-HD-Ultra Control Rod**

### **4.3 Nuclear Design**

This section describes the design bases and functional requirements used in the nuclear design of the fuel, core, and reactivity control system that complies with the requirements of CNSC REGDOC-2.5.2, Revision 1, Section 8.1 (Reference 4.1-1). The Nuclear Design Report for BWRX-300 Equilibrium Cycle, NEDC-33985P (Reference 4.3-1) provides detailed results for the reference equilibrium cycle.

#### **4.3.1 Design Bases**

The design bases require the plant to operate while meeting all safety requirements:

1. The reactivity bases prevent an uncontrolled positive reactivity excursion and stipulate subcriticality requirements
2. The overpower bases ensure the core will operate within fuel integrity limits

##### **4.3.1.1 Reactivity Feedback Bases**

Reactivity coefficients representing the differential changes in reactivity produced by differential changes in core conditions are used for calculating stability parameters and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler, moderator temperature, and the moderator void reactivity coefficient. Also associated with the BWR is a power reactivity coefficient. The power coefficient is a combination of the Doppler and moderator void reactivity coefficients in the power operating range and is not explicitly evaluated.

The fuel reactivity acceptance criteria are established in GESTAR (Reference 4.4-8) and each of the following fuel parameters must be negative throughout the life of the core:

- Doppler reactivity coefficient for all operating conditions
- Core moderator void reactivity coefficient resulting from boiling in the active flow channels for any operating conditions
- Moderator temperature coefficient for temperatures equal to or greater than hot standby
- Power coefficient, as determined by calculating the reactivity change resulting from an incremental power change from a steady-state base power level for all operating power levels above hot standby
- Net prompt reactivity feedback originating from prompt heating of the moderator and fuel for a super prompt critical reactivity insertion accident (e.g., control rod drop accident)

The Doppler coefficient, the moderator void coefficient and the moderator temperature coefficient of reactivity are negative for power operating conditions, thereby assuring negative reactivity feedback characteristics.

##### **4.3.1.2 Control Requirements (Shutdown Margins)**

It is required that the core can be shut down and remain subcritical with the minimum margin as specified in the plant Operating Limits and Conditions (Chapter 16), with the highest worth control rod, or rod pair associated with the common Hydraulic Control Unit (HCU) postulated stuck in the full out position.

#### 4.3.1.3 Control Requirements (Overpower Bases)

It is required that the core operate within an absolute power and power distribution envelope to ensure fuel integrity is maintained. This is controlled through two nuclear design basis parameters: the Maximum Linear Heat Generation Rate (MLHGR) Limit and Minimum Critical Power Ratio (MCPR). The MCPR and Linear Heat Generation Rate (LHGR) limit are determined with 95% confidence that the fuel does not exceed Operational Limits and Conditions during AOOs. These constraints must then be met under normal operating conditions.

Explicit Maximum LHGR Limit and MCPR parameter definitions are provided as follows:

**Maximum Linear Heat Generation Rate Limit:** The LHGR limit is the maximum allowable linear heat generation for each fuel rod in the bundle. The LHGR operating limit is bundle type-dependent and is a function of gadolinia content and exposure. The LHGR is monitored, and the fuel is not operated at MLHGR values greater than those found acceptable by the safety analysis under normal operating conditions. Under AOO conditions, including the maximum overpower condition, the calculated overpower is confirmed to neither cause fuel melting nor exceed the stress and strain limits as discussed in Subsection 4.2.3.

**Minimum Critical Power Ratio:** MCPR is the minimum CPR allowed for a given bundle type to avoid boiling transition. The CPR is a function of several important parameters: bundle power, bundle flow, rod power peaking distribution, and bundle mechanical design. The plant Operating Limit Minimum Critical Power Ratio (OLMCPR) is established by considering the limiting AOOs for each operating cycle. The OLMCPR is determined to avoid boiling transition for 99.9% of the rods during the limiting analyzed AOO transient discussed in Chapter 15, Subsection 15.5.3.

#### 4.3.2 Core Nuclear Design Description

The core is light-water moderated and fueled with low enriched uranium dioxide fuel assemblies. The use of light water as a moderator produces a neutron energy spectrum where fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The void reactivity feedback effect is an inherent safety feature of the BWR system. Any system change that increases reactor power, either locally or core-wide, produces additional steam voids and reduces power.

The reactor core is arranged as an upright cylinder containing fuel assemblies located within the core shroud. The coolant flows upward through the core. The reactor core includes fuel assemblies, control rods, and nuclear instrumentation. The arrangement of fuel assemblies/core loading map is the same as the forced flow Kernkraftwerk Mühleberg Nuclear Power Plant (KKM) BWR in Switzerland and is displayed in Figure 4.3-1.

The core nuclear design of any BWR core for an operating cycle is comprised of the following elements:

- Core arrangement, lattice type and fuel product line that combine to establish the detailed geometry of the reactor core
- The energy utilization plan that defines the energy requirements
- Core coolant hydraulics (e.g., core flow, pressure, and inlet temperature)
- Nuclear design (i.e., enrichment and burnable poison distribution) of the fresh fuel
- Core loading pattern of fresh and exposed fuel
- Planned control rod patterns during power operation

An equilibrium cycle is a reactor state in which the same fresh fuel assemblies are loaded into the same locations, the exposed fuel is shuffled to the same locations and the cycle is depleted in the same way until an equilibrium state is achieved producing essentially identical results (e.g., energy generated, power distributions, reactivity margins, etc.) cycle after cycle. An equilibrium cycle is the best representation of the core over the life of the reactor. A reference equilibrium core as shown in Figure 4.3-2 has been developed for safety analysis to demonstrate that the BWRX-300 conforms to all regulatory requirements with high confidence. The reference BWRX-300 equilibrium cycle has been selected to be an annual cycle (i.e., 12-month refueling interval) with high, albeit normal, discharge exposure. Alternate refueling intervals (e.g., 18, or 24 months) may be applied to the BWRX-300 and conforms to all regulatory requirements.

The reference equilibrium cycle is loaded with multiple fresh nuclear fuel bundle types of various enrichments and gadolinia burnable poison designs that satisfy a multitude of requirements and objectives, including to optimize the core burnup while maintain core performance. The core loading pattern, operating control rod patterns, and core performance results are described in the Nuclear Design Report, NEDC-33985P (Reference 4.3-1) and the nuclear design of the fresh fuel assemblies is provided Fuel Bundle Information Report (Reference 4.3-6). The BWRX-300 core configuration, including a representative in-core instrumentation design and rod patterns, are depicted in Figure 4.3-1 and Figure 4.3-3 respectively. Select nuclear design information and core performance results are depicted in Figure 4.3-4 through Figure 4.3-9.

#### **4.3.2.1 Refueling Interval**

BWRs operate on distinct refueling intervals. At the end of a normal operating cycle, after shutdown, the reactor is disassembled, then generally the lowest reactivity fuel is discharged and new reload fuel is inserted and the fuel is shuffled into the core configuration for the upcoming cycle. The reactor is then reassembled, and the startup of the next cycle can commence. This activity, from shutdown of any operating cycle to startup of the next, is termed the refueling outage. Inspections and maintenance are also typically performed during the refueling outage on intervals specific to each piece of equipment or component.

For BWRs, there is not a specific limit on the length of an operating cycle, or planned refueling interval; however, requirements governing inspection and surveillance frequency must be satisfied and the safety analyses that constitute the safety basis must encompass the planned operating cycle length. Cycle specific operating limits are established to span the planned operating cycle length and provision is made for cycle extension (e.g., power coast down) as specified by the reactor owner.

Any planned operating cycle length must:

1. Support requirements governing inspections of the reactor internals & equipment
2. Reside within evaluated space (i.e., the technical inputs to the safety analyses must envelope the planned operating cycle length)

The cycle specific operating limits are derived from AOO analysis of the actual core design. Any refueling interval that results in acceptable thermal operating limits that conform to the Specified Acceptable Fuel Design Limits (SAFDL) can be supported.

BWRs operate on approximately 12-, 18-, or 24-month refueling intervals; however, operating cycles longer than 24 months have been conducted as well as intermediate lengths. The reference BWRX-300 equilibrium core design to support safety and performance evaluations was established to be a 12-month cycle because it illustrates the highest degree of operational flexibility associated with excess thermal margins (e.g., Figure 4.3-7 shows ~30% excess thermal margin compared to a normal design target of MFLPD < 0.9), where MFLPD is the Maximum Fraction Limiting Power Density, which is equivalent to the maximum ratio of any LHGR in the core compared to its exposure dependent, LHGR limit).

The safety analyses that are generally affected by increasing the operating cycle length, along with areas that are part of the operating envelope, are denoted in Table 4.3-1.

#### **4.3.3 Core Nuclear Analytical Methods**

The analytical methods used in the design and analysis of the BWRX-300 core during all states of normal operation are summarized below and described in detail in NEDC-33939 "Steady State Nuclear Methods" (Reference 4.3-2).

##### **4.3.3.1 Steady-State**

The principal tools used in the steady-state nuclear core analysis are the three-dimensional (3-D) BWR Core Simulator PANAC11 and the two-dimensional lattice physics code TGBLA06. The BWR Core Simulator is a coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

Natural circulation flow is determined by the Transient Reactor Analysis Code General Electric (TRACG) described in NEDE-32176P (Reference 4.3-3). The associated core flow is used as an input to the core simulator.

The simulator computes core reactivity, power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, and burnable poisons as described in the nuclear libraries and other constitutive state variables. The simulator is used to calculate reactivity variations through the cycle, shutdown margins and compliance with thermal limits (i.e., LHGR Limit and MCPR).

PANAC11 and the associated nuclear libraries produced by TGBLA06 have undergone extensive validation by comparing calculated results with alternate methods, end-of-cycle gamma scan data, and operating reactor data. PANAC11 is a USNRC approved method used for production core design, licensing analysis, and core exposure tracking for the BWR fleet. The exposure tracking process provides the opportunity for continuous comparison and validation of the nuclear methods against operating data.

Validation of the adequacy of PANAC11 and associated compliance with CNSC guidance is demonstrated in NEDC-33939P (Reference 4.3-2).

##### **4.3.3.2 Reactivity Coefficient**

The lattice physics and 3D core simulator are used in computing the change in neutron multiplication (i.e., reactivity inserted) caused by a change in state (e.g., change in fuel temperature, in-channel void fraction, etc.) when determining the reactivity coefficients.

#### **4.3.4 Core Nuclear Design Evaluation**

##### **4.3.4.1 Reactivity Coefficient**

The safety analysis methods are based on system and core models that include an explicit representation of core space-time kinetics. Therefore, the reactivity coefficients are not directly used in the safety analysis methods but are useful in the general understanding and discussion of core response to perturbations.

The reactivity coefficients evaluation is performed as part of new fuel design development assuring consistency with safety objectives. The GNF2 results are documented in NEDC-33270P, GNF2 Advantage Generic Compliance with GESTAR II, which concludes that all the criteria defined in GESTAR II have been met for the GNF2 fuel design (Reference 4.2-1).

##### **4.3.4.2 Reactivity Variation**

The excess reactivity needed to deliver the target cycle energy while maintaining rated thermal power is controlled by the control rod system supplemented by fuel rods containing a burnable absorber. When applied to any specific fuel cycle, these integral fuel burnable absorber rods are used to provide partial control of the excess reactivity available during power operation. The burnable absorber lowers the reactivity of fresh fuel and is designed to be largely depleted by the end of the first cycle of operation. Control rods are used during the cycle to compensate for the remaining hot excess reactivity and reactivity changes due to burnup. Control rods may also be used to control the power distribution. The burnable absorber design is established such that the remaining hot excess reactivity is consistent with target control patterns during steady-state power generation. The hot excess reactivity and associated control rod patterns for the reference BWRX-300 equilibrium cycle are documented in the Nuclear Design Report for BWRX-300 Equilibrium Cycle. Examples of typical control rod patterns are presented in Figure 4.3-3.

##### **4.3.4.2.1 Core Reactivity Effects**

###### **Control Reactivity**

Neutron absorbing control rods are the primary means to control reactivity in transient and accident analyses. During events that result in relatively fast positive reactivity feedback, control rods are inserted rapidly using stored hydraulic energy. This is referred to as a scram. During events that result in relatively slow positive reactivity feedback and do not require a reactor “scram”, the FMCRDs that are operated using electric motors can be used for slower control rod insertion, (see Chapter 4, Subsection 4.5.1). Reactivity feedback mechanism is key for some Postulated Initiating Event groups (see Table 15.2-1) in which rods are withdrawn in error. Control reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

###### **Doppler Reactivity**

Doppler reactivity is a reactivity feedback mechanism in BWR transient, and accident analyses associated with changes in fuel temperature. Doppler reactivity is negative with an increase in fuel temperature and becomes more important as the fuel temperature continues to increase. In Deterministic Safety Analysis (DSA) where reactivity feedback is important, doppler reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.



### **Void Reactivity**

Void reactivity is an important reactivity feedback mechanism in BWR transient and accident analyses. The void reactivity feedback is always negative and is typically stronger (more negative void coefficient) at the end of an operating cycle. It is also stronger for reload cores versus an initial reactor core that includes only fresh fuel. This reactivity feedback mechanism is the dominant feedback for some Postulated Initiating Event groups. Table 15.2-1 focuses on events initiated from conditions of normal power operation. In the DSA, where it is important, void reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response analyzed using TRACG, the primary DSA computer code (see Subsection 15.5.1.2) that complies with CSA 286.7 on computer code quality assurance.

### **Moderator Temperature Reactivity**

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature. The value of this coefficient is important during the startup of a BWR. During power operation, the coefficient is not important, because the moderator is boiling, and primarily remains at the saturation temperature corresponding to the operating pressure. In addition, moderator density changes caused by boiling are much larger than changes from moderator temperature changes and therefore, mask any effects. Moderator temperature feedback is accounted for in the DSA analysis of startup conditions when coolant temperature is key to the event response, and there is no significant voiding in the core. Once core boiling/voiding begins, void reactivity feedback becomes dominant.

### **Boron Reactivity**

For the BWRX-300, boron reactivity insertion (see Chapter 15, Section 15B), is only needed for long-term shutdown in very low probability events, where the hydraulic, and electric motors fail to insert a sufficient number of control rods.

### **Xenon Reactivity**

Xenon reactivity feedback is not typically accounted for during events in DSA because the rate of change of reactivity is slow. The effects of xenon are accounted for in analyses of shutdown margin. Xenon reactivity impacts are also considered during core design and monitoring.

Also refer to Chapter 15, Subsection 15.2.4.1 for discussion regarding the effects of core reactivity on AOOs and DBAs.

#### **4.3.4.3 Shutdown Margin**

The minimum cold shutdown margin for the reference BWRX-300 equilibrium core is presented in Figure 4.3-9 and documented in the Nuclear Design Report, NEDC-33985P (Reference 4.3-1).

#### **4.3.4.4 Thermal Limit**

The margins to thermal limits for the reference BWRX-300 equilibrium core are presented in Figure 4.3-7 through Figure 4.3-9 and documented in the Nuclear Design Report (Reference 4.3-1).

#### **4.3.4.5 Xenon Stability**

BWRs are not susceptible to xenon oscillations. The xenon stability evaluation has been demonstrated by:

- No observed xenon instabilities in operating BWRs
- Special tests conducted on operating BWRs forcing the reactor into xenon instability demonstrate that xenon transients are highly damped by the large negative moderator void feedback
- Simulation calculations

All these indicators demonstrate that xenon transients are highly damped in a BWR due to the large negative moderator void feedback. Moreover, the BWRX-300 reactor core, fueled with GNF2, is evaluated to be less susceptible to xenon oscillations as compared to KKM (that never experienced an oscillation mode) as the void fraction is expected to be higher and the corresponding void reactivity coefficient more negative.

#### **4.3.4.6 Thermal-Hydraulic Stability**

The most limiting stability condition in the BWRX-300 normal operating region is at rated power/flow condition. The BWRX-300 core remains stable throughout the entire operating domain. Refer to Subsection 4.4.8 for a discussion of thermal-hydraulic stability.

#### **4.3.4.7 Load Following Operation**

Core power maneuvering is a normal occurrence for BWRs in response to several motivations (e.g., periodic surveillance testing, managing equipment performance degradation, control rod exchanges to reduce fuel duty, low electricity demand, etc.). The load follow operation involves a rapid power increase or reduction. The thermal limits established for the fuel are developed to assure conformance to regulatory requirements during all modes of operation, including load following. Additionally, BWR Operating Guidelines have been developed to assist in mitigating low frequency Stress Corrosion Cracking Pellet Clad Interaction and are encoded into the core monitoring function such that predictions may be performed in advance of power maneuvers.

The fuel rod thermal-mechanical operating limits described in Section 4.2 have been established to be applicable to all modes of operation, including frequent power cycles associated with load following. The principal performance evaluation affected by frequent fuel rod power cycling pertains to material fatigue. The GNF2 fuel rod thermal-mechanical suite of analyses includes a fatigue evaluation assuming frequent power cycling and is discussed in the GNF2 Fuel Rod Thermal Mechanical Design Report, (Reference 4.2.11). Note that fuel rod performance degradation associated with material fatigue has not been encountered in BWR operation.

The thermal-hydraulic limits established for every operating cycle (e.g., the OLMCPR) are developed to assure the Fuel Cladding Integrity Safety Limit described in Section 4.4 is maintained during all conditions of power operation above the Low Power Thermal Limit. The OLMCPR is developed including provision for AOOs that initiate when the core is operating at off-rated power & flow conditions and are applicable load following.

During periods of frequent core power maneuvering, transient xenon affects the core reactivity and power distribution. The effects of transient xenon are predicted by the core monitoring function prior to performing the core power maneuver and control rod adjustments are applied to maintain the core power at specified levels.

#### **4.3.5 Changes from Previous Reactor Design**

The BWRX-300 reactor core operates via natural circulation flow and is considered a composite design comprised of the most desirable features developed and applied to the BWR fleet. The key features are:

1. There are 240 fuel assemblies arranged identically to the KKM reactor core.
2. The lattice type is the N-lattice that originated with the ABWR. The N-lattice provides additional moderator volume in the intra-assembly bypass gap as compared to earlier lattice types.
3. The core flow results from natural circulation and the nominal bundle flow during power operation is lower than forced circulation reactors.
4. The core average power density is low compared to most forced circulation BWRs and approximately 20% lower than KKM (i.e., 870 MWth vs. 1097 MWth).
5. The CRDs are fine motion that were developed for the ABWR.
6. GTs are used in-lieu of the traversing in-core probe system for neutron instrument calibration.
7. The reference control rod type is the most modern commercially available control rod – the Ultra-HD.

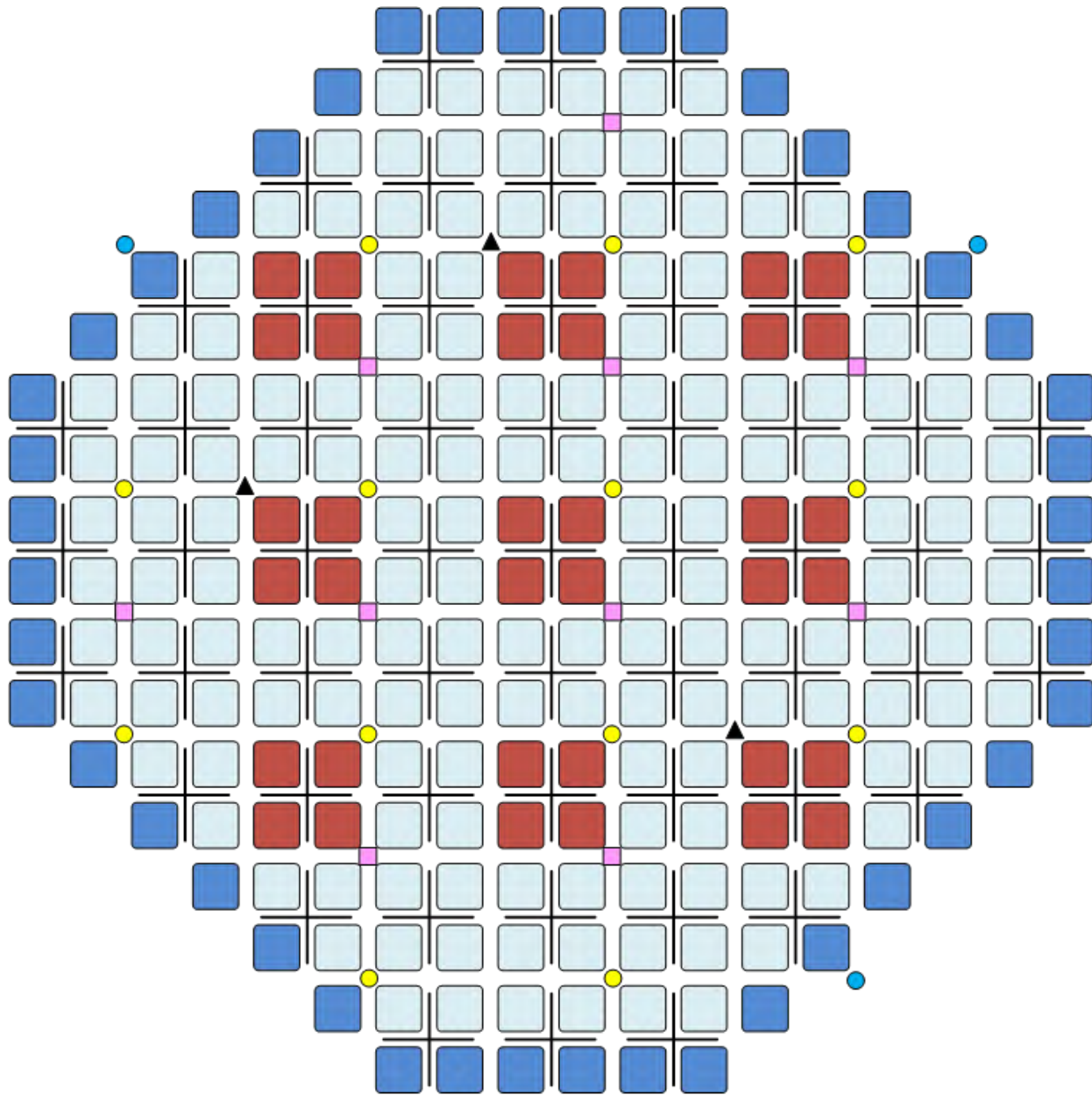
While the exact configuration of the BWRX-300 reactor core is new, the configuration is similar to the BWR operating fleet, and the performance of all principal aspects have been proven in fleet application. Any differences from KKM are encompassed within the current approved nuclear methods.








#### **4.3.6 References**

- 4.3-1 NEDC-33985P, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Equilibrium 12 Month Cycle Nuclear Design Report,” GE-Hitachi Nuclear Energy Americas, LLC.
- 4.3-2 NEDC-33939P, “BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology,” GE-Hitachi Nuclear Energy Americas, LLC.
- 4.3-3 NEDE-32176, “TRACG Model Description,” GE-Hitachi Nuclear Energy Americas, LLC.
- 4.3-4 NEDC-32082P SH 0001, “BWR Steady State Thermal Hydraulic Methodology (ISCOR),” GE-Hitachi Nuclear Energy Americas, LLC.
- 4.3-5 NEDC-33986P, “BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle,” GE-Hitachi Nuclear Energy Americas, LLC.

**Table 4.3-1: Cycle Independent Safety Analyses and Operating Envelope Evaluations  
Potentially Affected by Operating cycle Length**

Potentially Impacted Area	Evaluation Scope
Transient & stability analyses	Changes to plant & fuel specific, cycle-independent safety analyses
Core isotopic inventory and dose consequence	Changes to accident analyses' dose consequence and equipment qualification
Accident performance analysis	Changes to margin for fuel & reactor limits under accident conditions
Fluence impacts	Changes on the projected lifetime fluence for the RPV and reactor internals
Decay heat impacts	Changes to decay heat assumptions within fuel pool cooling, fire event analyses, and containment analyses
Technical specification surveillance intervals and instrument setpoints	Align to the outage schedule



- |   |                                 |   |                                |
|---|---------------------------------|---|--------------------------------|
|  | - Interior Assembly Locations   |  | - LPRM + GT Locations (13)     |
|  | - Interior Control Cells (A2)   |  | - WRNM Locations (10)          |
|  | - Peripheral Assembly Locations |  | - Level Sensor Locations (3)   |
|   |                                 |  | - Neutron Source Locations (3) |

**Figure 4.3-1: Core Map and Preliminary Instrumentation Layout**

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

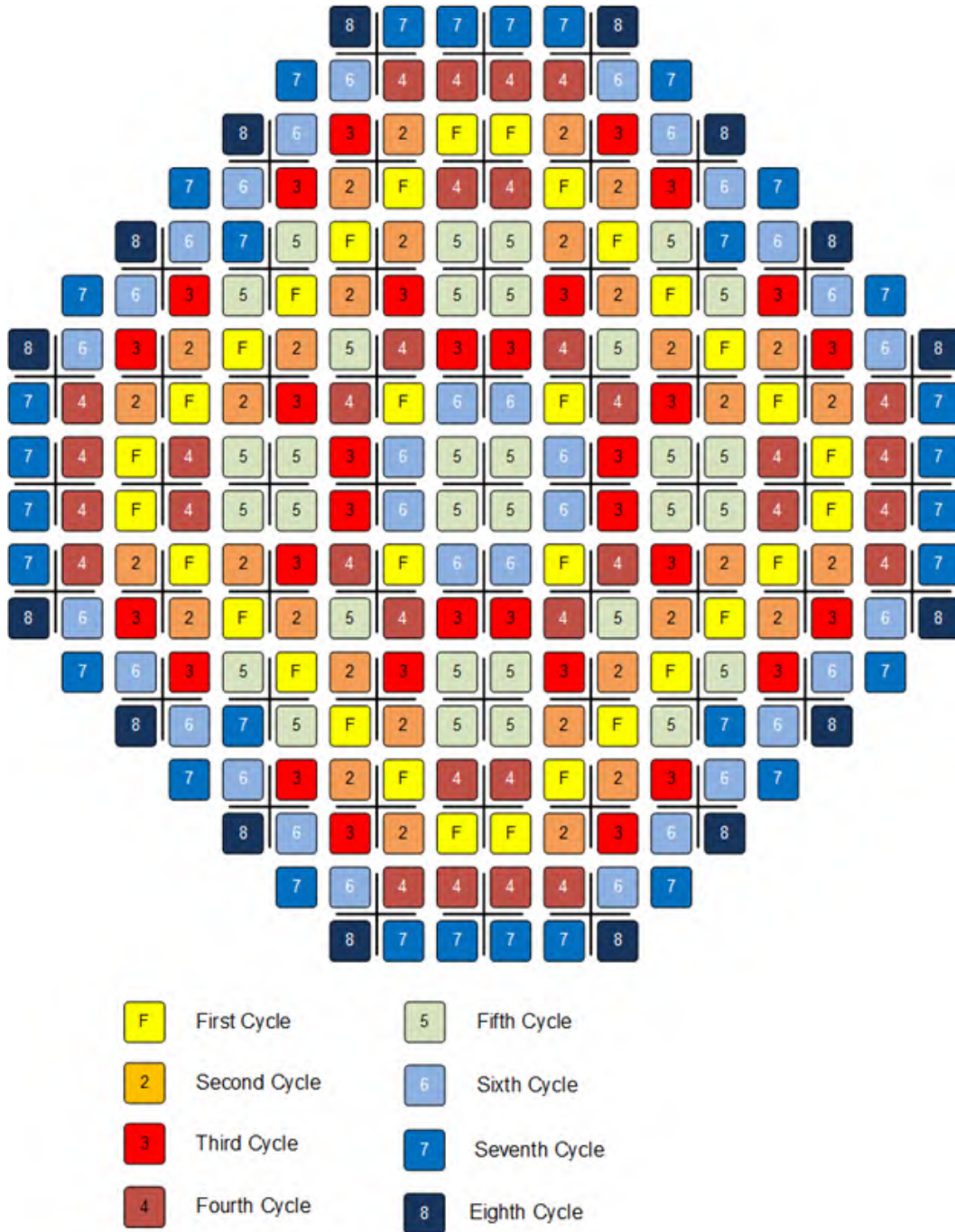
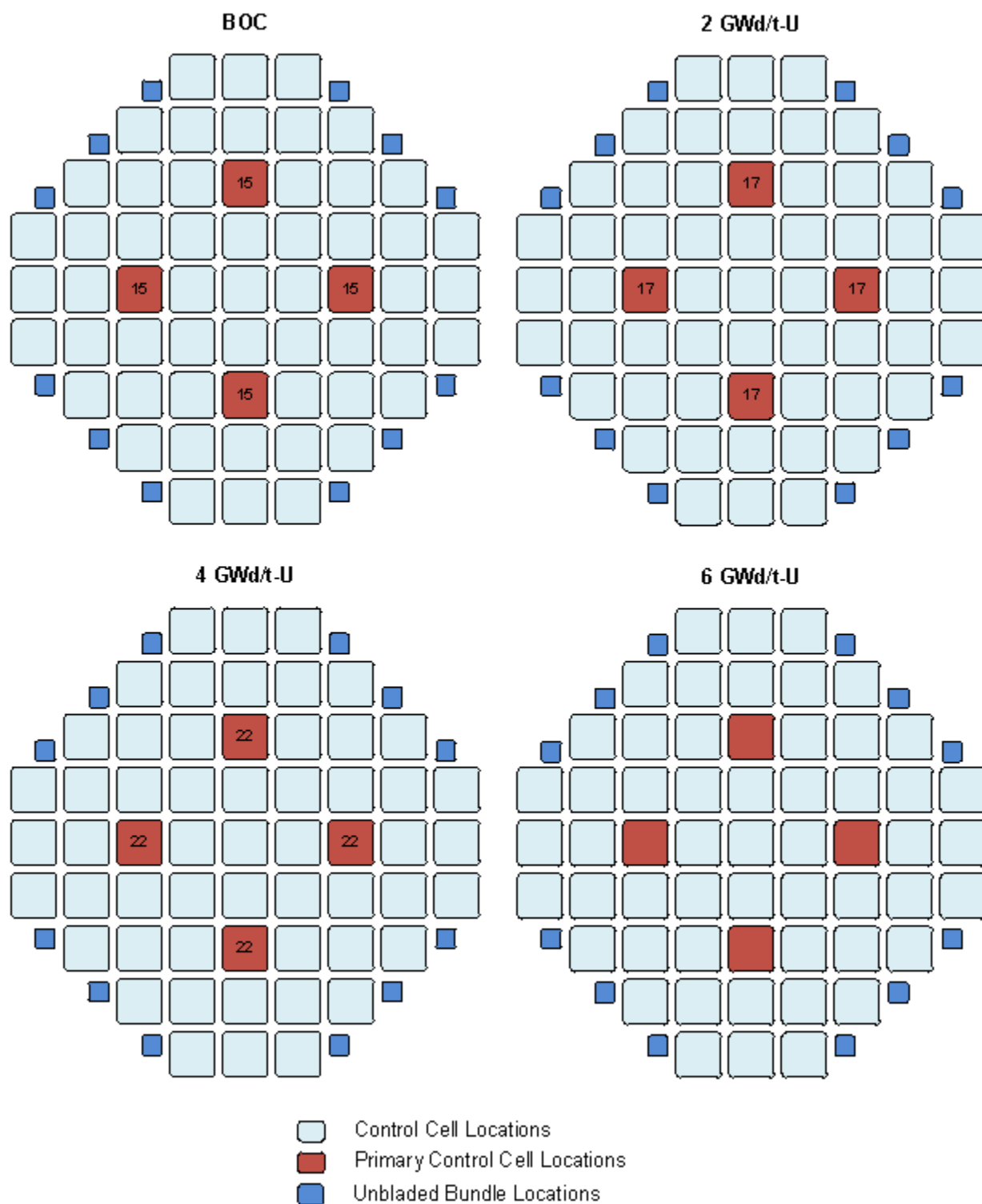
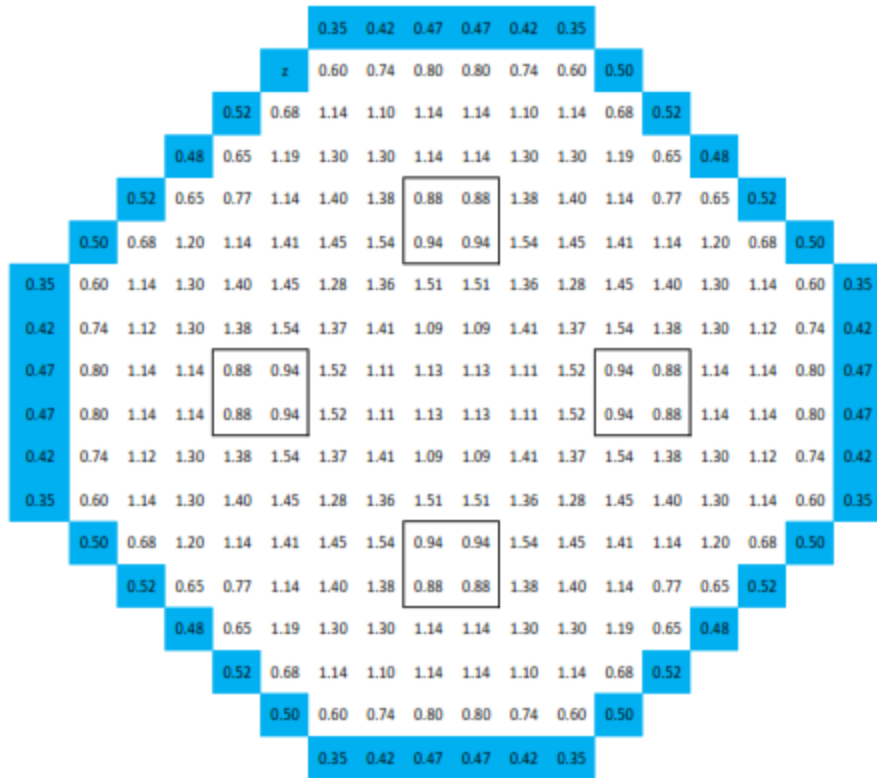


Figure 4.3-2: Core Loading Map – Reference Equilibrium Cycle



**Figure 4.3-3: Representative Control Rod Patterns**

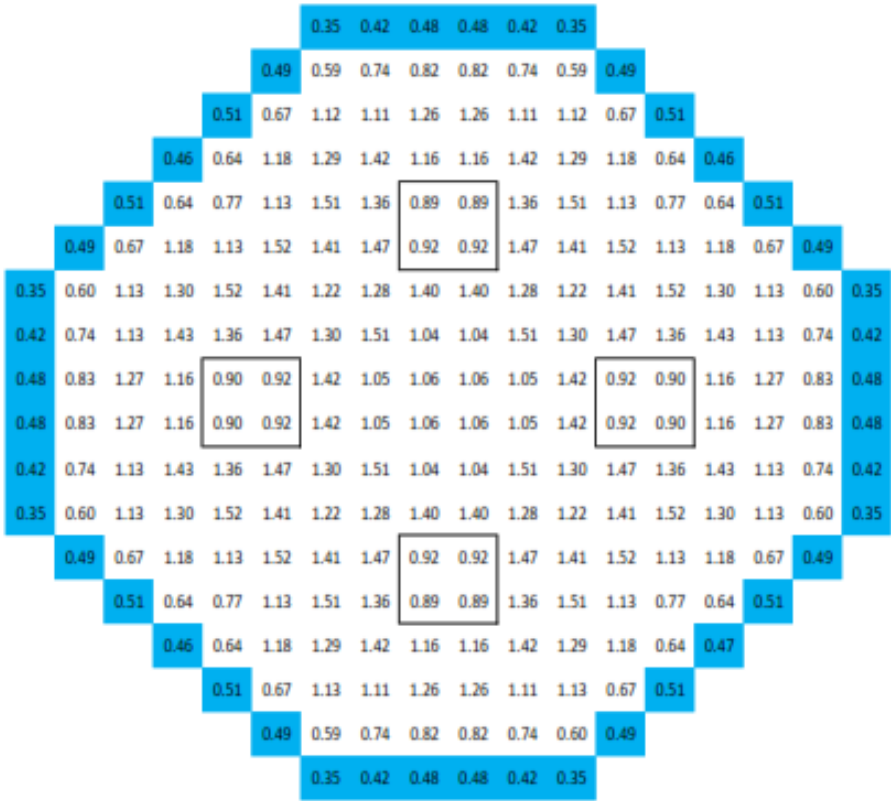
NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 4.3-4a: Peak to Average Bundle Power Ratio at Beginning of Cycle  
(Average Bundle Power – 3.625 MW)**

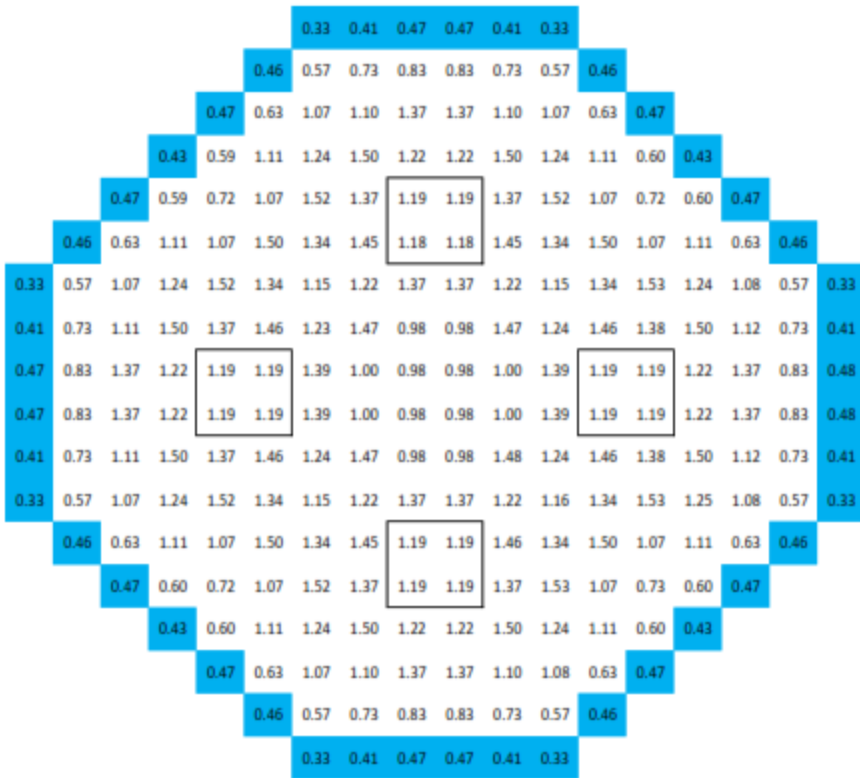


NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

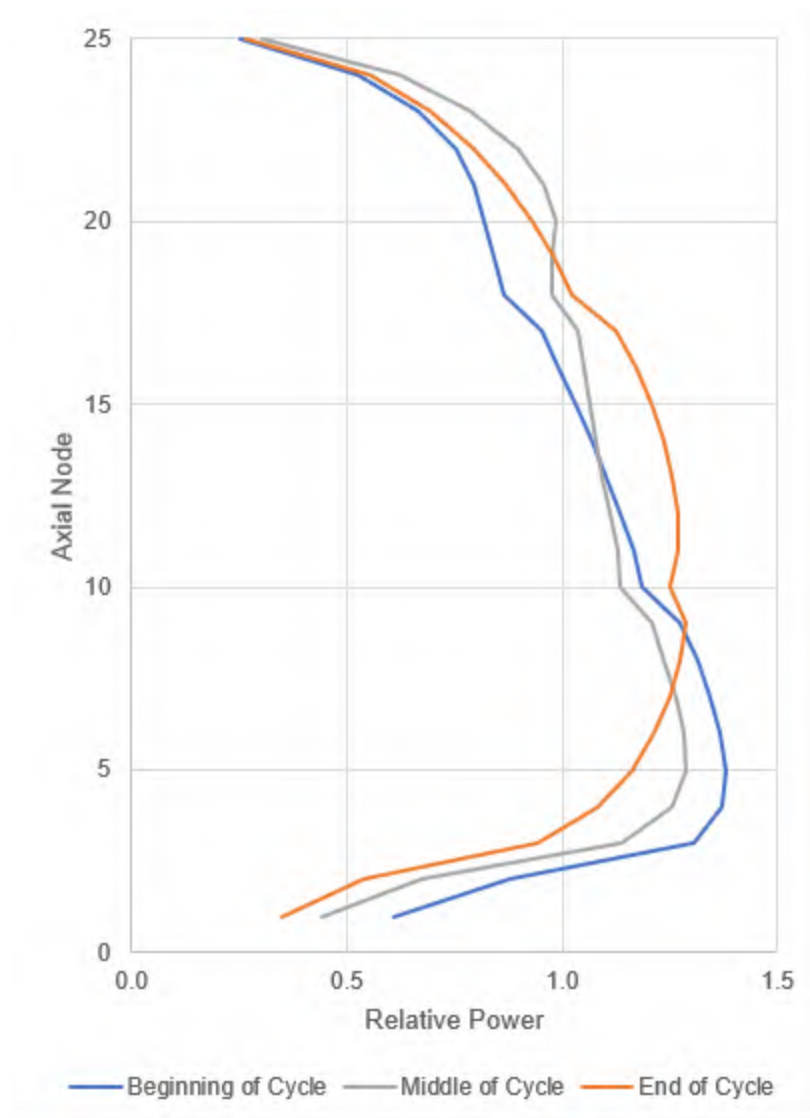


**Figure 4.3-4b: Peak to Average Bundle Power Ratio at Middle of Cycle  
(Average Bundle Power – 3.625 MW)**

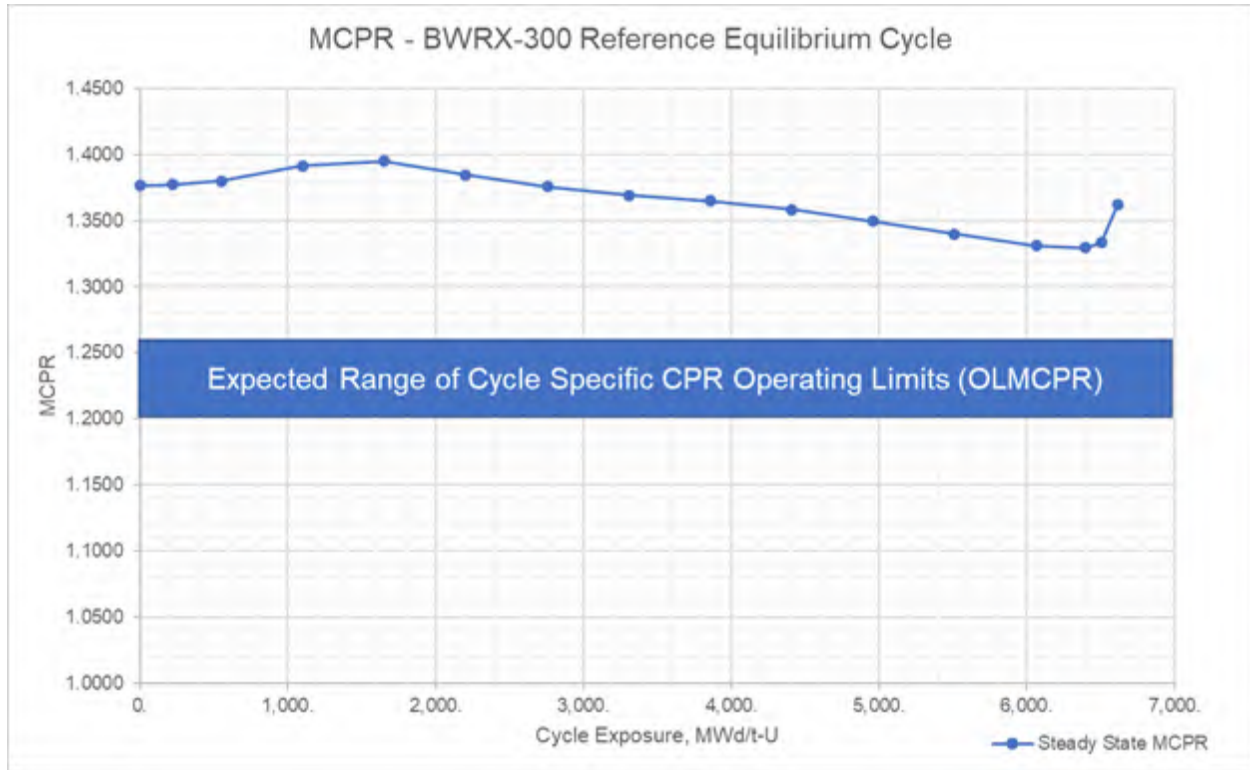
NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 4.3-4c: Peak to Average Bundle Power Ratio at End of Cycle  
(Average Bundle Power – 3.625 MW)**

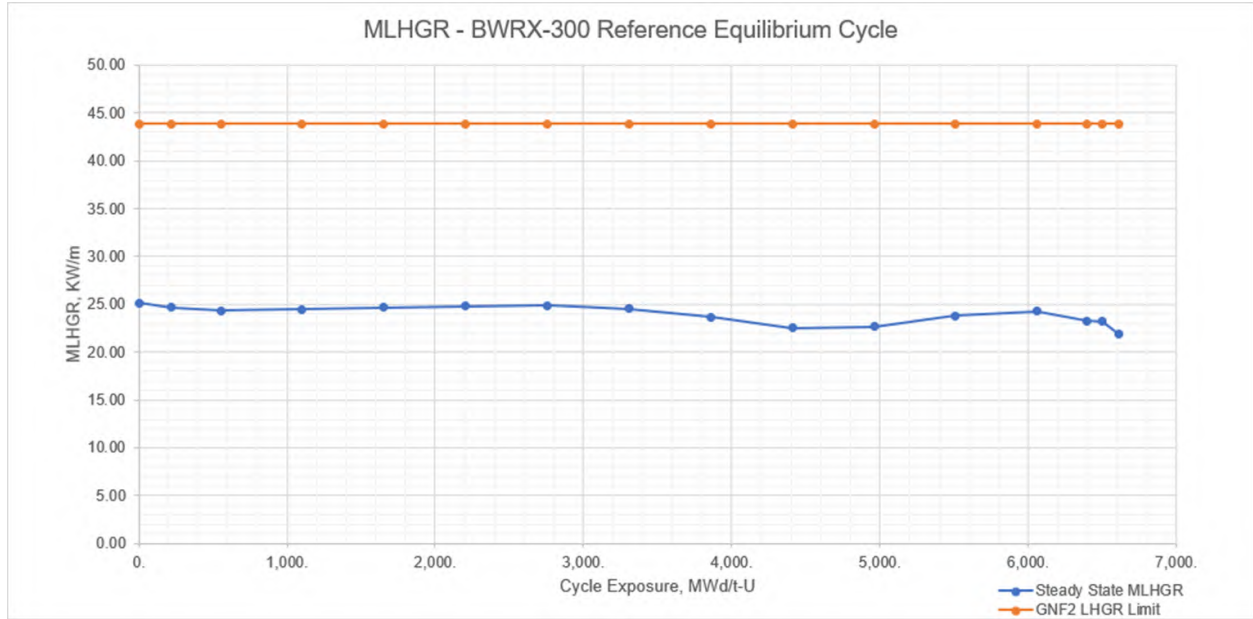


**Figure 4.3-5: Axial Power Shapes**

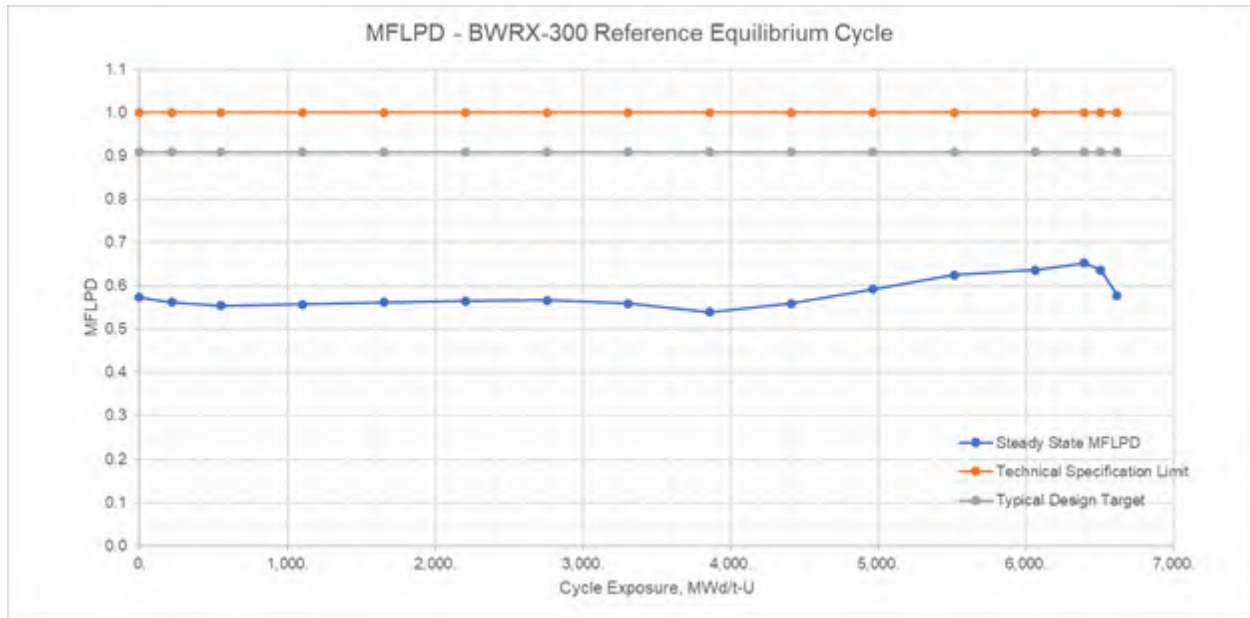


**Figure 4.3-6: Reference Equilibrium Cycle Thermal Margin – Minimum Critical Power Ratio**

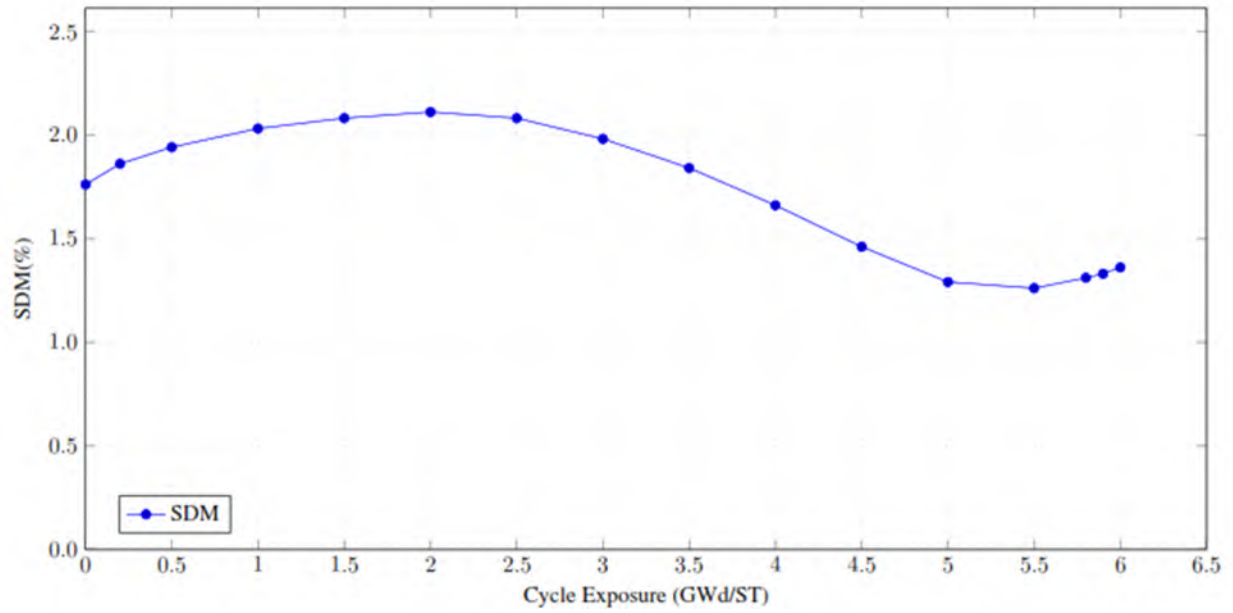
NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 4.3-7: Reference Equilibrium Cycle Thermal Margin – Maximum Linear Heat Generation Rate**



**Figure 4.3-8: Reference Equilibrium Cycle Thermal Margin – Maximum Fraction Limiting Power Density**



**Figure 4.3-9: Shutdown Margin vs Cycle Exposure**

#### **4.4 Thermal and Hydraulic Design Basis**

The thermal and hydraulic design of the reactor core provides adequate heat transfer from the fuel to the heat removal system, NBS, normal Shutdown Cooling System (SDC), or Isolation Condenser System during normal operation and AOOs.

##### **4.4.1 Design Bases**

Loss of fuel rod cladding integrity is not expected during normal reactor operation and AOOs. To satisfy this requirement, the following design bases have been established for the thermal and hydraulic design of the reactor core.

The core thermal-hydraulic design establishes thermal-hydraulic operating limits used in assuring safety margin in accordance with CNSC REGDOC-2.5.2, Version 1 (Reference 4.1-1). The core thermal-hydraulic stability performance addresses CNSC REGDOC-2.5.2 requirements in Subsection 4.4.8 Thermal-Hydraulic Stability.

Margin to the SAFDL is maintained during normal steady state operation when the MCPR is greater than the required OLMCPR and the MLHGR is maintained below the maximum LHGR limit(s). The steady state OLMCPR and Thermal Mechanical Operating Limit are established for the most limiting AOO and are analyzed in Chapter 15, Subsection 15.5.3 including uncertainties that provide reasonable assurance that no fuel damage results during AOOs.

##### **4.4.1.1 Critical Power**

The objective for normal operation and AOOs is maintaining nucleate boiling and precluding boiling transition.

The figure of merit confirming compliance with this objective is the CPR. The CPR is the ratio of the bundle power where at least one fuel rod point within the assembly experiences the onset of boiling transition to the operating bundle power. A calculated CPR of 1.0 corresponds to the best estimate value for the onset of boiling transition as determined by the product-specific GEXL correlation (e.g., GEXL17 for GNF2).

CPR limits are specified for maintaining adequate margin to the onset of the boiling transition. Adequate margin is defined to be a 95 % probability at a 95 % confidence level that no fuel rods are susceptible to boiling transition. These limits are calculated based on the three-step process defined in the sections that follow.

##### **4.4.1.1.1 Fuel Cladding Integrity Safety Limit**

The Fuel Cladding Integrity Safety Limit is calculated so that no significant fuel damage occurs during normal operation and AOOs on a cycle-independent basis. The Fuel Cladding Integrity Safety Limit is defined as the MCPR that ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition. This limit is also referred to as Safety Limit MCPR or  $MCPR_{95/95}$ . The value is dependent only on the fuel design and establishes a lower limit for the cycle-specific  $MCPR_{99.9\%}$  value.

##### **4.4.1.1.2 $MCPR_{99.9\%}$**

The  $MCPR_{99.9\%}$  is determined on a cycle-specific basis to support the determination of the Operating Limit MCPR (OLMCPR). The  $MCPR_{99.9\%}$  ensures that 99.9 % of the fuel rods in the core are not susceptible to boiling transition when considering the nuclear core design, plant system uncertainties, manufacturing uncertainties, and calculational uncertainties.



#### **4.4.1.1.3 Operating Limit Minimum Critical Power Ratio**

A cycle-specific OLMCPR provides adequate assurance that the  $MCPR_{99.9\%}$  is not exceeded during normal operation and AOOs. By operating with the MCPR at or above the OLMCPR, the Fuel Cladding Integrity Safety Limit for that plant is not exceeded during normal operation and AOOs. This operating limit is obtained by combining the maximum Delta Critical Power Ratio Over Initial Critical Power Ratio ( $\Delta CPR/ICPR$ ) (the change in CPR through the transient divided by the Initial CPR (ICPR)) value for the most limiting AOO and the  $MCPR_{99.9\%}$ . The significance of the  $\Delta CPR/ICPR$  is that it measures the transient response. The maximum  $\Delta CPR/ICPR$  from the AOOs is used in combination with the  $MCPR_{99.9\%}$  to establish the OLMCPR on a cycle-specific basis.

#### **4.4.1.2 Maximum LHGR**

The Maximum LHGR (MLHGR) bases are described in Subsection 4.3.1. Thermal mechanical operating limits ensure margin to design limits for circumferential cladding strain and centerline fuel temperature. The adequacy of LHGR limits is evaluated for the most severe AOOs providing assurance that no fuel damage results during these postulated events.

#### **4.4.1.3 Void Fraction Distribution**

The void fraction in a BWR fuel bundle has a strong effect on the neutron flux and power (or fission rate) distribution. Accurate prediction of the void fraction is important for evaluating the performance of the BWR reactor and fuel. The void fraction is evaluated using correlations based on the characteristic dimensions of the fuel bundle and hydraulic properties of the two-phase flow in the fuel bundle.

#### **4.4.1.4 Core Pressure Drop and Hydraulic Loads**

An accurate model of core pressure drop is essential for modeling natural circulation flow, fuel and core inlet flow, and hydraulic loads for input to other evaluations.

#### **4.4.1.5 Core Coolant Flow Distribution**

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions are calculated. The core coolant flow distribution forms the basis for predicting steady-state and transient MCPR and void fraction.

#### **4.4.1.6 Fuel Heat Transfer**

Engineering models in both steady state and transient analysis tools predict heat transfer between fuel pellet, cladding gap, cladding, fuel rod surface and the coolant in the evaluation of core and fuel safety criteria.

#### **4.4.1.7 Summary of Design Bases**

Steady-state operating limits ensure the design bases are satisfied during normal operation and the most limiting AOO. These limits are determined using analytical methodologies to evaluate core flow distributions, void fraction distributions, and fuel heat transfer rates as a function of both cycle exposure and core state.

The limiting AOO effects are confirmed maintaining compliance with the design acceptance criteria for the fuel, the RPV, and containment set forth in Chapter 15, Subsection 15.3.1. The barriers maintain their integrity and function.

#### **4.4.2 Thermal and Hydraulic Methods**

The analytical methods described in this section are qualified according to the provisions of American Society of Mechanical Engineers (ASME) NQA-1 which are equivalent to the provisions of CSA N286.7 (Reference 4.4-1). These methods have been approved by the USNRC (the country of origin) and are generally accepted for use by other countries with operating BWRs (e.g., Mexico, Spain, Switzerland).

##### **4.4.2.1 Fuel Bundle Critical Power Methods**

The critical power is the fuel bundle thermal power at the onset of boiling transition. Maintaining the bundle power below the critical power during steady-state operation and AOOs precludes the onset of boiling transition and satisfies the SAFDL pertaining to heat transfer from the fuel to the coolant (i.e., the Fuel Cladding Integrity Safety Limit). The methods applied in determining the bundle critical power and the associated operating limits are described below.

###### **4.4.2.1.1 Fuel Bundle Critical Power Performance**

The bundle critical power performance methodology was originally described in NEDO-10958-A (Reference 4.4-2). This original methodology evolved into the current form of the correlation, i.e., the GEXL correlation. The GEXL correlation is a critical quality and boiling length correlation used to predict the occurrence of boiling transition in BWR fuel. Each fuel bundle design has a specific set of correlation coefficients developed from full-scale test data. The specific GEXL correlation applied in analyzing GNF2 for all BWR types, including BWRX-300, is designated GEXL17 NEDC-33292 (Reference 4.4-3). The GEXL17 correlation application range established for fleetwide application envelopes the hydraulic conditions that the BWRX-300 experiences during normal operation and AOOs.

###### **4.4.2.1.2 Fuel Cladding Integrity Safety Limit Method**

The fuel cladding integrity safety limit, named the  $MCPR_{95/95}$ , ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition using a limit that is derived from comparing the predicted critical power to the experimental data for a specific fuel bundle design. The Experimental CPR is defined as the ratio of the calculated critical power as determined by the GEXL correlation to the experimental critical power.

Each experimental data point has a predicted value and associated Experimental CPR. The Experimental CPR is evaluated for all the points in the dataset resulting in a probability distribution. The Experimental CPR probability distribution serves as the basis for the correlation uncertainty. Thus, for a given critical power correlation, a limit that bounds 95% of a correlation's Experimental CPR distribution at a 95% confidence level is determined and set as the  $MCPR_{95/95}$ .

The determination of the  $MCPR_{95/95}$  is further described in TSTF-564 (Reference 4.4-9).

###### **4.4.2.1.3 $MCPR_{99.9\%}$**

The cycle specific  $MCPR_{99.9\%}$  limit is established as described in NEDC-33939, Appendix C (Reference 4.3-2).

###### **4.4.2.1.4 OLMCPR Calculation Method**

BWRX-300 AOOs are analyzed using the TRACG model described in NEDE-32176P (Reference 4.3-3). The application of TRACG to AOO analyses is described in TRACG Application for BWRX-300, NEDO-32082 (Reference 4.3-4) along with the process to calculate the  $\Delta CPR/ICPR$ .

This OLMCPR is obtained by combining the maximum  $\Delta\text{CPR}/\text{ICPR}$  value for the most limiting AOO and the  $\text{MCPR}_{99.9\%}$ . The OLMCPR is specified in one of two ways. It can be conservatively defined as the most limiting combination of  $\Delta\text{CPR}/\text{ICPR}$  and  $\text{MCPR}_{99.9\%}$  over an entire cycle, or it is defined on a cycle exposure-dependent basis using combinations of cycle exposure-dependent values of  $\Delta\text{CPR}/\text{ICPR}$  and/or  $\text{MCPR}_{99.9\%}$ . Chapter 15, Section 15.5 provides the analysis of limiting AOOs.

#### **4.4.2.2 Maximum Linear Heat Generation Rate Method**

The MLHGR methods are described in Subsection 4.3.1. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with the TRACG Application for BWRX-300 (Reference 4.3-4).

#### **4.4.2.3 Void Fraction Distribution Methods**

The empirical correlations used for the calculating void fraction are the GEH void fraction correlation used in the 3D core simulator, steady-state thermal hydraulic calculations, and the correlations for the interfacial shear used in TRACG. The TRACG void fraction model is described in NEDE-32176P (Reference 4.3-3). The core simulator model is described in NEDC-33939P (Reference 4.3.2).

#### **4.4.2.4 Core Pressure Drop and Hydraulic Loads Methods**

The total bundle pressure drop is defined as the sum of four components: friction, elevation, acceleration, and local losses. In these models, the bundle is also divided into control volumes where the four components of total pressure drop are evaluated separately. This allows capturing the effects on pressure drop of axially variable geometry parameters such as flow area, hydraulic diameter, wetted/heated perimeters, heat flux, and spacer elevations. The hydraulic diameter is defined as four times the axial flow area divided by the wetted perimeter, at any axial location and includes the fuel rod, channel inner wall, and water rod perimeters. The geometry of heated surfaces consists of the number of fuel rods and the fuel rod diameter in a fuel assembly. For fuel assembly types with partial length rods, the number of partial length rods and the associated length(s) are also accounted for in defining fuel assembly hydraulic diameter.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P (Reference 4.3-3). The TRACG hydraulic formulation for core pressure drop is identical to the model used in the core design analysis except for the acceleration pressure drop component. The models used in the core design analysis are described in NEDE-333939 (Reference 4.3-2). The fuel design specific loss coefficients and assembly pressure drop models are developed from and confirmed by data from full scale testing of prototypical assemblies spanning the range of hydraulic conditions where hydraulic models are applied. The adequacy of the pressure drop model applied to GNF2 fuel is summarized in GNF2 pressure drop characteristics and is inclusive of the BWRX-300 operating conditions in GNF2 Pressure Drop Calculations, NEDO-32082 (Reference 4.4-4).

Hydraulic loads are determined based on the reactor internal pressure differences. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

#### **4.4.2.5 Core Coolant Flow Distribution Methods**

The core coolant flow distribution methods used in TRACG are described in NEDE-32176P, Chapters 6 and 7 (Reference 4.3-3). TRACG treats all fuel channels as one-dimensional (axial) components, but the vessel is modeled as a three-dimensional component. Hence, the pressure drop across two planes in the vessel is the same at all radial and azimuthal locations if the geometry of the components in the vicinity of these planes has radial and azimuthal symmetry. Otherwise, this pressure differential displays some (locally) radial and azimuthal non-uniformity.

The flow distribution to the fuel assemblies and bypass flow paths in the core simulator model is calculated assuming the pressure drop across all fuel assemblies and bypass flow paths is the same. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses described above. The core inlet flow is an input to the core simulator. The value used in core design analysis is determined based on the TRACG prediction of the natural circulation core inlet flow. In operation, the core monitoring function determines core inlet flow based on plant instrumentation discussed in Chapter 7, Subsection 7.3.3.2.

The bypass flow methodology is described in NEDE-32176P (Reference 4.3-3), Subsection 7.5.1. The same methodology is used in the core simulator model.

#### **4.4.2.6 Fuel Heat Transfer Methods**

The Jens-Lottes heat transfer correlation is used to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling as described in (Reference 4.4-5). For the single-phase convection or liquid region, the Dittus-Boelter correlation is used. The methodology for fuel cladding, gap and pellet heat transfer is described in Subsection 4.4.1.1.4 and (References 4.2-5 through 4.2-8).

### **4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Core**

Several metrics of interest for the BWRX-300 are compared against a typical BWR/6 plant and the ABWR in Table 4.4-1.

#### **4.4.3.1 Reactor Coolant System**

The Reactor Coolant System is described in Chapter 5. The BWRX-300 reactor coolant system is shown in Figure 4.1-1. The BWRX-300 thermal hydraulic design is similar to operating BWRs except that it does not require recirculation pumps or associated coolant piping. Circulation of the reactor coolant through the BWRX-300 core is accomplished via natural circulation. Natural circulation is enabled mainly by the addition of a tall chimney between the top of the core at the top guide plate of the core to the bottom of the chimney head and steam separator assembly. The natural circulation flow rate depends on the difference in water density between the regions. The core flow varies according to the power level because fluid density changes with the power level. Therefore, a core power-flow map reduces to a single line and there is no active control of the core flow at any given power level, as shown in Figure 4.4-2.

#### **4.4.3.2 Core Hydraulics**

Accurate prediction of bundle flow and power distributions is important in the calculation of margin to the thermal limits of each fuel bundle. Pressure drop characteristics are included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.

Because of the channeled configuration of BWR fuel assemblies, there is no assembly-to-assembly cross flow inside the core. The only issue of hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its impact on margin to thermal limits (i.e., MCPR or MLHGR). The coupled thermal-hydraulic-nuclear analyses are performed each cycle to determine fuel bundle flow and power distribution. The analyses use the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions applied to all fuel bundles. The margin to thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

The flow distribution to the fuel assemblies and bypass flow paths is calculated using various pressure drop models that include friction loss coefficients, local loss coefficients, two-phase multipliers, and void-quality correlations. These models are developed from pressure drop data with a best-fit basis. Pressure drop measurements made in operating reactors, confirm that the total measured and calculated core pressure drops agree. This information is collected normally as part of core management activities and its purpose is to identify anomalous behavior. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support casting and/or peripheral fuel support (experiencing a pressure loss) where some of the flow exits through the fit-up between the fuel support and lower tie plate and through the lower tie plate holes into the bypass flow region. All fuel bundles have lower tie plate holes. Most of the flow continues through the lower tie plate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tie plate into the bypass region.

The unique GNF2 fuel assembly hydraulic characteristics include the inlet orifice, lower tie plate, upper tie plate, spacers, water rod, and various leakage paths. The hydraulic characteristics of these components flow paths have been developed and confirmed by test comparisons. These unique GNF2 hydraulic characteristics are used in all analysis models and methods where the fuel assembly hydraulics are needed.

The analytical methods used in the analysis of the BWRX-300 reactor are Global Nuclear Fuels Americas LLC (GNF A) standard codes in use throughout the industry and licenced in other jurisdictions.

Flow pressure drop characteristics shall be included in plant cycle specific analyses for the calculation of the Operating Limit MCPR

#### **4.4.3.3 Core Response in Transient and Accident Analysis**

Core and fuel response to postulated AOOs and accident events is explicitly modeled via the TRACG computer code employing three-dimensional kinetics that is consistent with the BWR Core Simulator PANAC11. A description of the core response during off-normal condition is provided in Chapter 15, Subsection 15.2.4 The results of AOO and DBA analyses are presented in Chapter 15, Subsections 15.5.3 and 15.5.4.

#### **4.4.3.4 Physics and Reactivity Coefficients Including Effects or Power Coefficient of Reactivity**

The principal reactivity coefficient for the BWRX-300, and any BWR, is the void coefficient, which is the dominant constituent of the power coefficient of reactivity and required to be negative under all reactor states.

Power control to ensure compliance with LHGR limits, including aspects of loss of reactivity control, is based on the use of cruciform shaped control rods (which are sometimes referred to as blades), like other BWR types, for reactor thermal power, power distribution, and control of the BWRX-300 core. Fast acting shutdown capability is provided using stored hydraulic energy HCUs as a diverse motive force relative to the FMCRDs, as described in Section 4.6. The Core Monitoring function (Chapter 7, Section 7.3) in concert with reactor operators ensure compliance with requirements implemented by the thermal limits described in Subsections 4.3.1 (MLHGR) and 4.4 (MCPR) during power operation.

#### **4.4.3.5 Operation Limits and Conditions for Core, Core Instrumentation and Control, and Nuclear Fuel**

The core design strategy for various refueling cycle intervals is based on management of thermal limit margins for fuel design.

Plant OLMCPR is established by considering the limiting AOOs for each operating cycle. This may be calculated as a function of exposure. Reports are normally prepared for reactor operators in implementing the Cycle Management Report (CMR) requirements for each refueling cycle interval.

For each new fuel design, the applicability of generic or country-specific MCPR analyses are confirmed for each operating cycle, or a plant-specific analysis is performed.

#### **4.4.3.6 Load Following Characteristics**

The thermal-hydraulic limits established for every operating cycle (e.g., the OLMCPR) are developed to assure the Fuel Cladding Integrity Safety Limit described in Section 4.4 is maintained during all conditions of power operation above the Low Power Thermal Limit. The OLMCPR is developed including provision for AOOs that initiate when the core is operating at off-rated power & flow conditions and are applicable load following.

The BWRX-300 is a natural circulation BWR. The power-flow map depicted in Figure 4.4-2 illustrates that the core flow is fairly constant in the normal range of core thermal powers associated with load following. As the reactor power is the principal parameter that varies, and reactor coolant flow is fairly constant, compliance to thermal limits is readily maintained during periods of off-rated power operation. Further discussion of load following is provided in Subsection 4.3.4.7.

### **4.4.4 Thermal and Hydraulic Evaluation**

#### **4.4.4.1 Critical Power Evaluations**

Compliance to representative steady-state MCPR operating limits is demonstrated for a typical simulation of an equilibrium cycle described in Section 4.3, Core Nuclear Design. The typical OLMCPR evaluation process is outlined in the sections that follow.

##### ***4.4.4.1.1 Fuel Cladding Integrity Safety Limit Evaluation***

The GNF2 Fuel Cladding Integrity Safety Limit value is 1.07 as per TSTF-564-A (Reference 4.4-9).

#### **4.4.4.1.2 *MCPR<sub>99.9%</sub>***

The MCPR<sub>99.9</sub> is evaluated for a specific core design with uncertainties documented in (Reference 4.3-2), NEDC-33939. The MCPR<sub>99.9</sub> limit is computed on a cycle-specific basis and reported in a cycle specific Core Operating Limits Report (COLR).

#### **4.4.4.1.3 *Minimum Critical Power Ratio Operating Limit Evaluation***

The MCPR Operating Limit is computed on a cycle-specific basis and reported in a cycle specific COLR. The expected range of cycle-specific OLMCPRs for the reference BWRX-300 equilibrium cycle is depicted in Figure 4.3-6.

#### **4.4.4.2 Maximum Linear Heat Generation Rate Evaluations**

Compliance to steady-state MLHGR limits is demonstrated for the reference equilibrium cycle in Section 4.3 and the Nuclear Design Report (Reference 4.3-1). The AOO analysis for the reference equilibrium cycle is documented in Chapter 15, Subsection 15.5.3. Compliance to design limits for circumferential cladding strain and centerline fuel temperature during AOO events are confirmed on a cycle-specific basis and associated limits are reported in the COLR.

#### **4.4.4.3 Void Fraction Distribution Evaluation**

The void fraction distribution is dependent upon the reactor state and varies throughout an operating cycle. The calculation of the void fraction distribution is integral to the TRACG methodology and the 3D core simulator, PANAC11. Representative values for the core average axial void fraction are depicted in Figure 4.4-4.

#### **4.4.4.4 Core Pressure Drop and Hydraulic Loads Evaluations**

The expected operating pressure for the BWRX-300 is within the qualification basis of the pressure drop methods. The MCPR<sub>99.9%</sub> calculation method also assumes pressure drop uncertainty.

#### **4.4.4.5 Core Coolant Flow Distribution Evaluation**

The core coolant flow distribution (i.e., the inlet flow to each fuel assembly) is determined by the coupled nuclear and thermal hydraulic steady-state methods in NEDC-33939 (Reference 4.3-2) and in the design and analysis of the reference core nuclear design summarized in Section 4.3.

#### **4.4.4.6 Fuel Heat Transfer Evaluations**

Fuel heat transfer evaluations are dependent upon the reactor state. The calculation of fuel heat transfer is integral to the TRACG methodology and the 3D core simulator, PANAC11.

#### **4.4.5 Evaluation of the Validity of Thermal and Hydraulic Design Techniques**

The thermal and hydraulic design technique comprises qualified analytical methods employed in developing a self-consistent set of design outcomes that conform to design bases. The TRACG method described in Subsection 4.4.2 demonstrates accurate model system performance in BWRs. The thermal hydraulic design bases evaluated for the BWRX-300 described in Subsection 4.4.1 are applicable and adequate to those established for the operating fleet. The BWRX-300 reactor core was developed from applying qualified analytical methods and the results demonstrate compliance to the design bases that are used in BWRs.

**4.4.6 References**

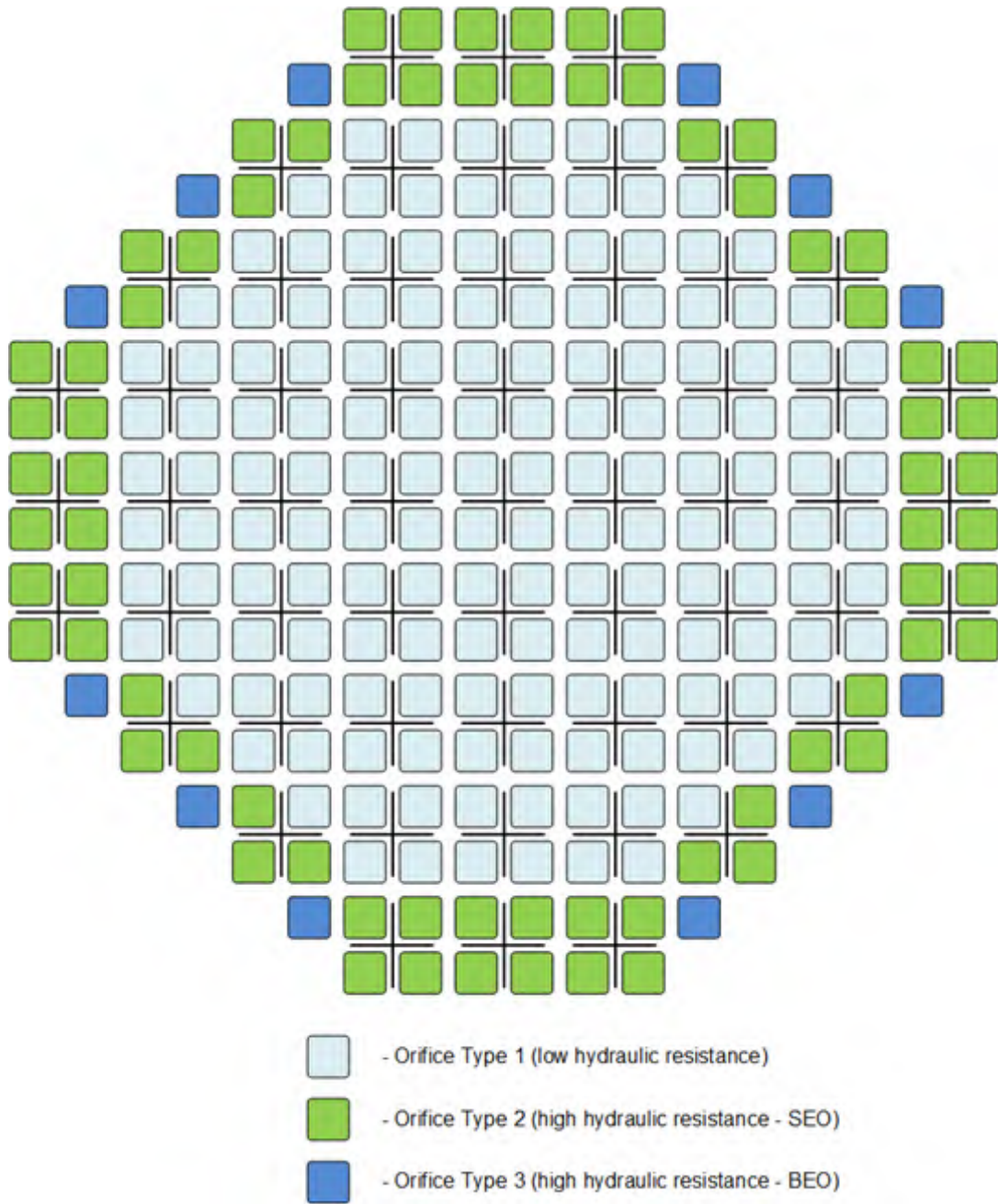
- 4.4-1 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 4.4-2 NEDO-10958-A SH 0001,"General Electric BWT Thermal Analysis Basis (GETAB) Data, Correlation and Design application, Licensing Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-3 NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-4 NEDC-33976P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Pressure Drop Characteristics," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-5 ANL-4627, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High-Pressure Water," Argonne National Laboratory.
- 4.4-6 NEDC-33083, "TRACG Application for ESBWR," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-7 NEDE-32177, "TRACG Qualification," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-8 NEDE-24011, "NEDE-24011-P-A-31, General Electric Standard Application for Reactor Fuel (GESTAR II)," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-9 TSTF-564-A, "Safety Limit MCPR," Technical Specifications Task Force.



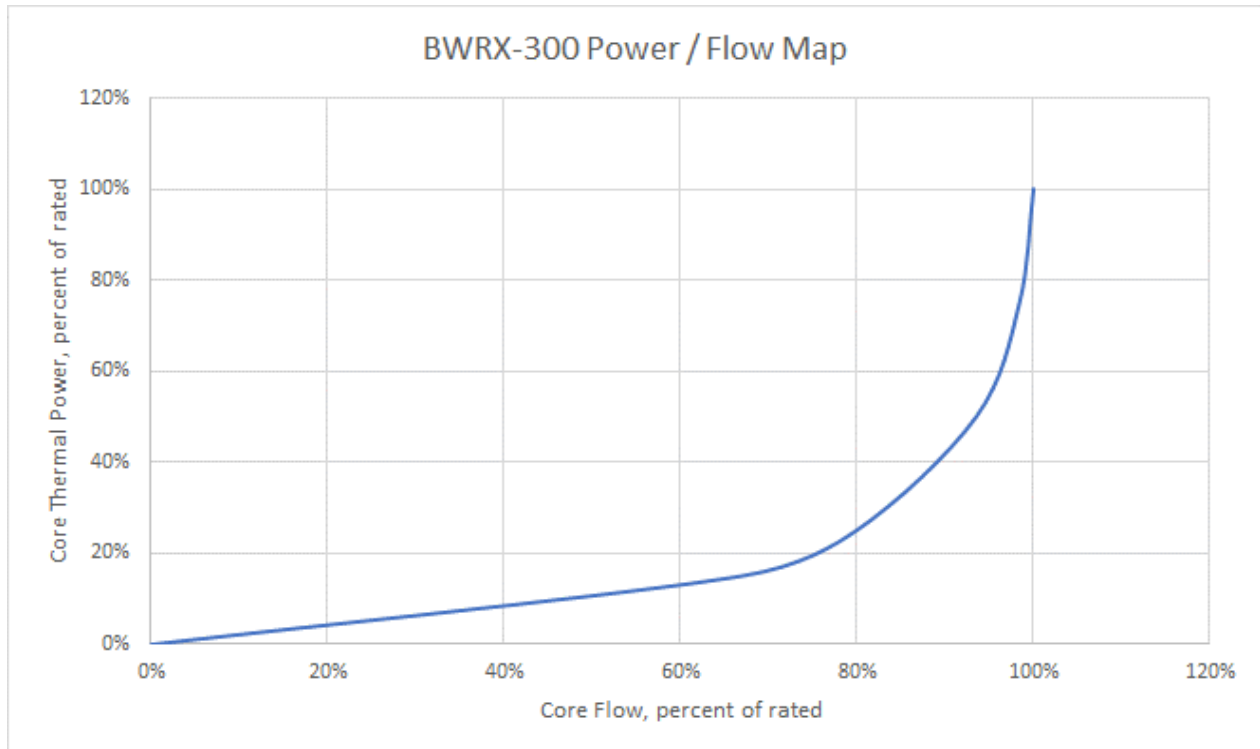
**Table 4.4-1: Typical Thermal–Hydraulic Design Characteristics of the Reactor Core**

General Operating Conditions	BWR/6	ABWR	BWRX-300
Reference design thermal output (MWt) <sup>(1)</sup>	3579	3926	870
Power level for engineered safety features (MWt)	3730	4005	887
Steam flow rate (kg/s)	1940	2122	503
Core coolant flow rate (kg/s)	13104	14502	1890
Feedwater flow rate (kg/s)	1936	2118	507
System pressure, nominal in steam dome (kPa)	7171	7171	7171
Coolant saturation temperature at core design pressure (°C)	288	288	288
Average power density (kW/L)	54.1	50.6	39.6
Core inlet enthalpy (kJ/kg)	1227	1227	1218
Core inlet temperature (°C)	278	278	269-272
Core maximum exit voids within assemblies (%)	79.0	75.1	90.0
Core average void fraction, active coolant	0.414	0.408	0.530
Total core pressure drop (kPa)	182.0	168.2	71.98
Core support plate pressure drop (kPa)	151.7	137.9	49.6

(1) 1MW = 1E6 J/sec



**Figure 4.4-1: BWRX-300 Core Inlet Orifice Type Arrangement**



**Figure 4.4-2: BWRX-300 Power / Flow Map**

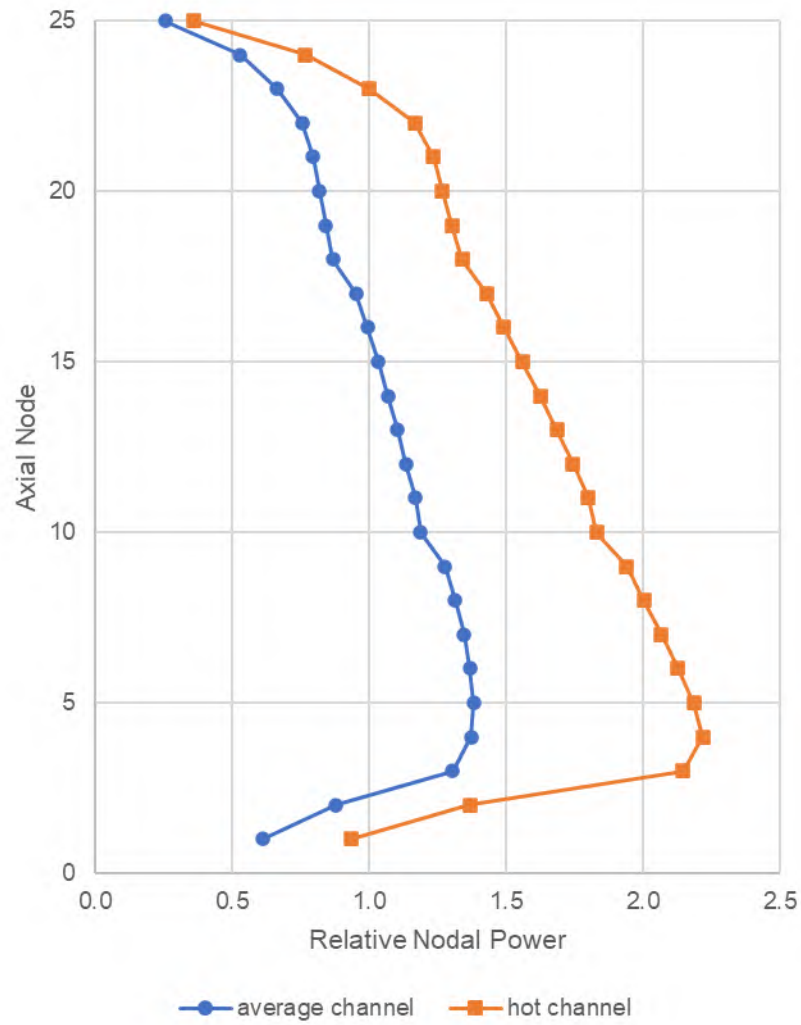
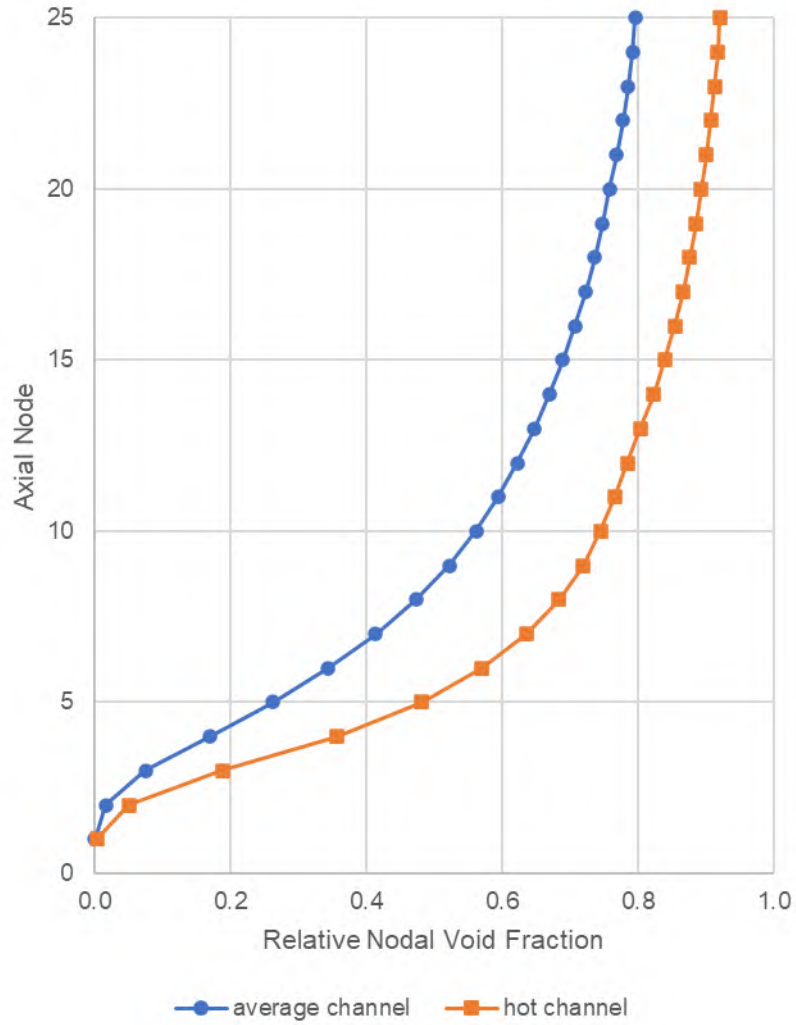


Figure 4.4-3: Relative Power for Analyzed Node (hot channel and average channel)



**Figure 4.4-4: Relative Void Fraction for Analyzed Node (hot channel and average channel)**

#### **4.5 Reactor Internals Material**

The materials used in the reactor internals are described in Chapter 5, Subsection 5.2.5, and Table 5.2-3.

The materials comprising the fuel and control rod assemblies are described in 4.2, and associated references.

The reactor coolant pressure boundary components of the FMCRD, consists of the lower component housing and the upper component's middle flange. These components are made with 300 series stainless steel materials compatible with the reactor coolant in accordance with ASME Code, Section III. The installation bolts used to attach the middle flange and lower component to the CRD housing are low alloy steel material.

##### **4.5.1 References**

None.

## **4.6 Design of Reactivity Control Systems**

Reactivity Controls consist of:

- Control rods (Section 4.2) and Control Rod Drive Systems
- Supplementary reactivity control in the form of gadolinia-urania fuel rods (Section 4.2.2)

This section describes the CRD system design, including the design bases, functions, system configuration & operation.

### **4.6.1 Control Rod Drive Design Bases**

The CRD system provides the primary means of reactivity control during normal, abnormal and accident conditions. The system design basis includes two diverse motive forces for the CRD insertion (scram) using high pressure water from the HCU's, and control rod insertion using the FMCRD motor. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component break or other malfunction. Complete system functionality is described below in Subsection 4.6.2.

### **4.6.2 Description**

The CRD system includes three major elements:

- Electro-hydraulic FMCRD mechanisms
- HCU
- CRD hydraulic subsystem

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods.

The hydraulic power required for scram is provided by high pressure water stored in the individual HCU's. Each HCU contains a scram accumulator (nitrogen-water), charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCU's provide a flow path for purge water to the associated FMCRDs.

The CRD hydraulic subsystem provides clean, demineralized water that is used to charge the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of pressurized water for purging the SDC pumps and filling the Nuclear Boiler System (NBS) reactor water level reference leg instrument lines.

The CRD system performs the following functions:

1. Control changes in core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the Rod Control and Information System (RC&IS).
2. Provides positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from RC&IS.
3. Provides the ability to position large groups of rods simultaneously in response to control signals from RC&IS.
4. Supplies rod status and rod position data for rod pattern control, performance monitoring, operator display, and scram time testing to the RC&IS.
5. Supplies purge water flow to the FMCRDs for cooling and foreign material exclusion.
6. Supplies water for sampling to the Process Radiation Monitoring System (PRM).

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

7. Supplies a continuous flow of water to the NBS to keep the reactor water level reference leg instrument lines filled.
8. Supplies charging water to fill and pressurize the HCU accumulators.
9. Supplies a makeup source for the RPV pressure test.
10. The FMCRD maintains Reactor Coolant Pressure Boundary (RCPB) integrity.
11. The FMCRD prevents rod ejection from the core due to a break in the drive mechanism pressure boundary or a failure of the attached scram line.
12. Supplies purge water for the SDC pumps.
13. Performs the scram follow function following any hydraulic scram signal.
14. Provides the capability for hydraulic scram in response to signals from the Anticipatory Trip System.
15. Provides the capability for Selected Control Rod Rapid Insertion (SCRR) in response to signals from the Anticipatory Trip System.
16. In conjunction with RC&IS and the Emergency Rod Insertion Control Panel (ERICP), the CRD system blocks the withdrawal of control rods in response to signals from the Advanced Thermal Limit Monitor (ATLM) and the Multichannel Rod Block Monitor (MRBM).
17. In conjunction with RC&IS, the CRD system detects separation of the hollow piston from the ball nut or separation between the control rod blade and the hollow piston and imposes a rod withdrawal block.
18. Provides the capability for hydraulic scram upon receiving a scram signal from the SC1 I&C system.
19. Provides the capability for hydraulic scram that is independent from the SC1 hydraulic scram upon receiving the scram signal from the Diverse Protection System.
20. Performs the Alternate Rod Insertion (ARI) function to provide an alternate means of hydraulic scram.
21. Provides the capability for fast run-in of the FMCRDs upon receiving a signal from the FMCRD controllers with input from the ERICPs.
22. Provides maximum available flow to the RPV (as part of DL4b when a signal is received from the I&C systems).

The design bases and further discussion of both the RC&IS and the hydraulic scram function as well as their control interfaces with the CRD system, are presented in Chapter 7.

The CRD system separates the SC1 equipment from the rest of the system. The FMCRDs are mounted to the reactor vessel bottom head inside containment. The HCUs are housed in four rooms located directly outside of containment in the Reactor Building. These rooms are arranged around the periphery of the containment wall. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates containment.

The balance of the hydraulic system equipment (e.g., pumps, valves, filters) is physically separated from the HCUs by barriers and is connected to the HCUs via the purge water header, scram accumulator charging water header and scram air header.

The system interfaces of the CRD system are shown in Figure 4.6-1.



#### **4.6.2.1 Fine Motion Control Rod Drive Mechanism**

The FMCRD used for positioning the control rod in the reactor core is an electro-hydraulic actuated mechanism (Figure 4.6-2 and Figure 4.6-3). An electric motor-driven ball nut and ball screw assembly positions the drive at both nominal increments and continuously over its entire range at a nominal speed. The FMCRDs also have the capability for motor-driven fast control rod insertion. Hydraulic pressure from the HCU is used for scrams. A single HCU powers the scram action of two FMCRDs, except for the FMCRD in the center of the core which has its own HCU. The FMCRD penetrates the bottom head of the RPV. The FMCRD does not interfere with refueling and is operative even when the head is removed from the RPV.

#### **4.6.2.2 FMCRD Components**

Simplified schematic of the FMCRD is provided in Figure 4.6-3. The basic elements of the FMCRD are as follows:

- Components required for electrical rod positioning or fine motion control (including the motor, brake release, ball screw, ball nut and hollow piston)
- Components required for hydraulic scram (including hollow piston and buffer)
- Components required for pressure integrity (including the middle flange, installation bolts and lower component)
- Rod position indication (position signal detectors)
- Reed position switches for scram surveillance and full-in indication
- Control rod separation detection devices (dual FMCRD separation switches)
- Bayonet coupling between the hollow piston and control rod
- Brake mechanism to prevent rod ejection in the event of a break in the FMCRD primary pressure boundary
- Ball check valve to prevent rod ejection in the event of a failure of the scram insert line
- Integral internal drive blowout support (to prevent drive blowout)
- Magnetic coupling

These features and functions of the FMCRD are described below.

#### **Components for Fine Motion Control**

The fine motion capability is achieved with a ball nut and ball screw arrangement driven by an electric motor. The ball nut is keyed to the guide tube (roller key) to prevent its rotation as it traverses axially as the ball screw rotates. A hollow piston rests on the ball nut and upward motion of the ball nut drives the control rod into the core. The weight of the control rod keeps the hollow piston and ball nut in contact during withdrawal.

The drive motor, located outside the pressure boundary, is magnetically coupled to the drive shaft located inside the pressure boundary. A splined coupling connects the drive shaft to the ball screw. The lower half of the splined coupling is keyed to the drive shaft and the upper half keyed to the ball screw. The tapered end of the drive shaft fits into a conical seat on the end of the ball screw to keep the two axially aligned. A drive shaft thrust bearing carries the entire weight of the control rod and drive internals.

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

The axially moving parts are centered and guided by radial rollers. The ball nut and bottom of the hollow piston includes radial rollers bearing against the guide tube. Radially adjustable rollers at both ends of the labyrinth seal keep the hollow piston precisely centered in this region.

A stationary guide supports the top of the rotating ball screw against the inside of the hollow piston. A hardened bushing provides the circumferential bearing between the rotating ball screw and stationary guide. Rollers of the guide engage with axial grooves in the hollow piston to prevent the guide from rotating with the ball screw.

### **Components for Scram, Scram Follow, and Run-In**

Opening of the scram valve in the HCU initiates the hydraulic scram action. High pressure water lifts the hollow piston off the ball nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball nut releases spring-loaded latches in the hollow piston that engage slots in the hollow piston guide tube. These latches support the control rod and hollow piston in the inserted position. Simultaneous with the initiation of any hydraulic scram, a scram follow signal is generated such that each FMCRD motor drives the ball nut upward to a position just below the fully inserted and latched hollow piston. With the ball nut below the hollow piston, the piston remains latched to maintain the control rod in the fully inserted position.

The automatic run-in of the ball nut using the electric motor drive following the hydraulic scram provides a diverse means of rod insertion as a backup to the accumulator scram. During motor run-in, the hollow piston remains in contact with the upper surface of the ball nut. This results in the weight of the hollow piston/control rod assembly being supported by the ball nut with the hollow piston latches remaining in the retracted position.

Table 4.6-2 shows the scram performance requirements for the CRD system in terms of the maximum elapsed time for each control rod to attain the listed scram position (percent insertion) after loss of signal to the scram solenoid pilot valves (time zero).

### **FMCRD Pressure Boundary**

The ASME Section III Class 1 RCPB components include the CRD housing (attached to the RPV), the FMCRD middle flange including the ball check valve, and the FMCRD lower component housing (which encloses the lower part of the drive). The middle flange is attached to the CRD housing by four threaded bolts. The lower housing is held to the middle flange and secured to the CRD housing by a separate set of eight main mounting bolts that become a part of the reactor pressure boundary. This arrangement permits removing the lower housing without disturbing the rest of the drive. Removing the lower housing transfers the weight of the drive line from the drive shaft to a seat in the middle flange. Both the ball screw and drive shaft are locked to prevent rotation while the two are separated.

The part of the drive inserted into the CRD housing is contained within the outer tube. The outer tube is the drive hydraulic scram pressure boundary, eliminating the need for designing the CRD housing for scram pressure. The outer tube is welded to the middle flange at the bottom and is attached at the top with the CRD blowout support, which bears against the CRD housing (refer to Figure 4.6-8). The blowout support and outer tube are attached by a slip-type connection that accounts for any slight variation in length between the drive and the CRD housing.

Purge water continually flows through the drive. The water enters through the ball check valve in the middle housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals discharge the high-pressure scram water into the RPV without adversely affecting the movement of the hollow piston.

### **Rod Position Indicator**

Control rod position indication is provided by the FMCRDs to the control system by a position detection system that consists of position detectors and position signal converters.

Each FMCRD provides two position detectors, one for each control system channel, in the form of signal detectors directly coupled to the motor shaft. This configuration provides continuous detection of rod position during normal operation.

### **Scram Position Indicator**

Scram position indication is provided by a series of magnetic reed switches to allow for measurement of adequate drive performance during scram. The magnetic switches are located at intermediate intervals over 60% of the drive stroke. They are mounted in a probe exterior to the CRD housing. A magnet in the hollow piston trips each reed switch in turn as it passes by.

As the bottom of the hollow piston contacts and enters the buffer, a magnet is lifted that operates a reed switch indicating scram completion. This continuous full-in indicating switch (shown on Figure 4.6-4) provides indication whenever the drive is at the full-in latched position or above. In the event of failure of the buffer mechanism a buffer contact reed switch is positioned above the full-in switch to provide failed buffer indication.

### **Control Rod Separation Detection**

Two redundant switches are provided to detect the separation of the hollow piston from the ball nut or between the control rod blade and the hollow piston (see Figure 4.6-5). The separation switches function to detect a detached control rod and cause a rod withdrawal block, thereby preventing a rod drop accident. Actuation of either switch also initiates an alarm in the Main Control Room (MCR).

During normal operation, the weight of the control rod and hollow piston resting on the ball nut causes the ball screw assembly to compress a spring on which the lower half of the splined coupling between the drive shaft and ball screw assembly rests (the lower half of the splined coupling is also known as the "weighing table"). When the hollow piston separates from the ball nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and ball screw assembly upward. This action causes a magnet in the weighing table to operate the reed switches located outside the lower housing.

### **Bayonet Couplings**

There are two bayonet couplings associated with the FMCRD. The first is the FMCRD/control rod guide tube/housing interface (see Figure 4.6-8). This bayonet locks the FMCRD and the base of the control rod guide tube to the CRD housing and functions to retain the guide tube during normal operation and dynamic loading events. The bayonet also holds the FMCRD against ejection in the event of a hypothetical failure of the CRD housing weld. The locating pin on the core plate that engages the flange of the control rod guide tube and the bolt pattern on the FMCRD/housing flange, assure proper orientation between the control rod guide tube and the FMCRD to confirm that the bayonet is properly engaged.

The second bayonet (Figure 4.6-6) is located between the control rod and FMCRD. The coupling spud at the top end of the hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45° rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

The FMCRD design allows the coupling integrity of this second bayonet to be checked by driving the ball nut down into an overtravel out position. After the weighing spring has raised the spindle assembly to the limit of its travel, further rotation of the spindle in the withdraw direction will drive the ball nut down away from the piston (assuming the coupling is engaged). If the hollow piston is not properly coupled to the control rod, the hollow piston will remain in contact with the ball nut and move with it to the overtravel position. A reed switch at the overtravel position will detect this movement of the hollow piston and provide alarm in the MCR.

### **FMCRD Brake and Ball Check Valve**

The FMCRD design incorporates an electromechanical brake (Figure 4.6-7) keyed to the motor shaft. The brake is normally engaged by passive spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RC&IS. Disengagement is caused by the energized magnetic force overcoming the spring load force. The braking torque and the magnetic coupling torque between the motor and the drive shaft are sufficient to prevent control rod ejection in the event of failure in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram) or motor run-in. The brake is a unidirectional mechanism that allows the motor to drive the control rod only in the insert direction if the brake is unintentionally engaged.

The electromechanical brake is located in the motor unit. The stationary spring-loaded disk and coil assembly is contained within the brake mounting bolted to the bottom of the motor unit top flange. The rotating disk is keyed to the motor unit output shaft.

A ball check valve is located in the middle flange of the drive at the scram inlet port. The check valve actuates to close the scram inlet port under conditions of reverse flow caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection.

### **Integral Internal Blowout Support**

The CRD blowout support (Figure 4.6-8) is designed to prevent ejection of the FMCRD and the attached control rod in the unlikely event of:

- A failure of the weld between the CRD housing and the stub tube penetration of the bottom head, or a failure through the housing along the fusion line of the weld
- A total failure of all flange bolts attaching the lower component flange and middle flange to the CRD housing flange

The internal CRD blowout support utilizes the FMCRD outer tube integral with support internal to the RPV to provide the anti-ejection support. The outer tube, which is welded to the drive middle flange, attaches by a bayonet lock to the control rod guide tube base. The core plate in turn supports the control rod guide tube.

With a CRD housing failure, the weight plus pressure load acting on the drive and housing would tend to eject the drive. In this event, the control rod guide tube base remains supported by the intact housing extension inside the RPV and the FMCRD remains supported in turn by its positive lock to the control rod guide tube base. Coolant leakage is restricted to the small annular area between the CRD outer tube and the inside of the CRD housing. In the event of a total failure of the weld itself, with the entire CRD housing intact, the housing would tend to be driven downward by the total weight plus RPV pressure. However, after the interconnected assembly of the housing, FMCRD and control rod guide tube move down a short distance, the flange at the top of the control rod guide tube contacts the core plate and stops further movement of the assembly. Since the FMCRD is positively locked to the control rod guide tube base, it cannot eject. In this case, the housing that bears on top of the blowout support is also prevented from leaving the

penetration. Coolant leakage for this condition is restricted to the small annular area between the outside of the CRD housing and the inside of the penetration stub tube.

The FMCRD design provides an anti-rotation device that engages when the lower component is removed for maintenance (Figure 4.6-11). This device prevents rotation of the ball screw and hence prevents control rod motion when the lower component is removed. The anti-rotation device consists of:

- The coupling piece on the bottom of the ball screw that engages with the lower component drive shaft
- The back seat of the middle flange

The coupling between the lower component drive shaft and ball screw is splined to permit removal of the lower housing. The underside of the coupling piece on the ball screw has a circumferentially splined surface that engages with a mating surface on the middle flange backseat when the ball screw is lowered during lower component removal. When engaged, ball screw rotation is prevented. In the unlikely event of total failure of all the drive flange bolts attaching the lower component flange and middle flange of the drive to the housing flange, the anti-rotation device will engage when the lower component falls. The middle flange/outer tube/CRD blowout support will be restrained by the control rod guide tube base bayonet coupling, thus preventing rod ejection.

### **Magnetic Coupling**

The magnetic coupling is located at the bottom of the lower component (Figure 4.6-3). It is employed to achieve leak-free operation of the FMCRD without seals. The magnetic coupling consists of an inner and an outer rotor. The inner rotor is located inside the lower component pressure boundary. The outer rotor is located outside the pressure boundary. Each rotor has permanent magnets mounted on it. As a result, the inner and outer rotors are locked together by the magnetic forces acting through the pressure boundary and work as a synchronous coupling. The outer rotor is coupled with the motor unit and driven by the motor such that the inner rotor follows the rotation of the outer rotor.

The magnetic coupling is designed so that its maximum coupling torque exceeds the maximum torque of the motor unit to prevent decoupling or slippage due to motor torque.

The magnetic coupling is designed to not have any undesirable effects on other magnetic sensitive subcomponents.

### **Materials of Construction**

The materials of construction for the FMCRD components which form part of the RCPB are discussed in Chapter 4, Section 4.5. For the non-pressure retaining FMCRD components, materials are selected for compatibility with the reactor coolant, wear resistance, corrosion resistance and material strength to ensure reliability and design life requirements are met in the BWRX-300 environment.

#### **4.6.2.2.1 Hydraulic Control Units**

Upon receipt of a scram signal, each HCU furnishes pressurized water for hydraulic scram to two FMCRD units (except for the FMCRD in the center of the core which has its own HCU). Additionally, each HCU provides the capability to adjust purge flow to the two drives. A test port is provided on the HCU for connection to a portable test station to allow for controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown. The check valves shown inside the HCU boundary function to close under system pressure, fluid flow and temperature conditions during scram. The check valves ensure that the water stored in the HCU

accumulator is delivered to the FMCRDs to accomplish the scram function. A simplified single line diagram of the HCU is provided in Figure 4.6-9. The major components of each HCU are provided below.

### **Scram Solenoid Valve Assembly**

The scram solenoid valve assembly consists of two solenoid valves which control the position of the scram valve. The solenoid valves are normally energized and closed. Upon loss of power to the solenoids, the valves open which vents air to open the scram valve. The assembly is designed so that the power must be removed from both solenoids before air pressure can be discharged from the scram valve operator. This prevents the inadvertent scram of the drives associated with a given HCU in the event of a failure of one of the valve solenoids.

### **Scram Valve**

The scram valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the MCR as soon as the valve starts to open.

### **Scram Accumulator**

The scram accumulator stores sufficient energy to fully insert two control rods at any anticipated reactor pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in case charging water header pressure is lost. During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Pressure sensors provide local and MCR nitrogen pressure indication. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the MCR. The alarm would prompt operator action to repressurize the nitrogen bottle using an external supply of nitrogen gas. To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A level sensor actuates an alarm in the MCR if water leaks past the piston barrier and collects in the accumulator instrumentation block.

### **Purging Panel**

The purging panel controls the purge water flow to the associated FMCRDs. Each panel has a restricting orifice in the purge water line to control the purge water flow rate. This orifice maintains the flow at a constant value while the drives are stationary. A bypass line containing a solenoid-operated valve is provided around this orifice. The valve is signaled to open and increase the purge water flow whenever either of the two associated FMCRDs is commanded to insert by the RC&IS. During FMCRD insertion cycles, the hollow piston moves upward, leaving an increased volume for water within the drive. Opening of the purge water makeup valve increases the purge flow to offset this volumetric increase and precludes the backflow of reactor water into the drive, thereby preventing long-term drive contamination.

#### ***4.6.2.2.2 Control Rod Drive Hydraulic Subsystem***

The CRD hydraulic subsystem supplies clean demineralized water to the following:

1. HCU accumulators for charging
2. FMCRDs for purge water
3. RPV for makeup
4. SDC pumps for seal purge water

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

5. NBS reactor water level reference leg instrument lines
6. PRM for sampling

The subsystem consists of two trains, each providing the required functions with the pumps, valves, filters, piping, and instrumentation described below. A simplified single line diagram for the CRD is provided in Figure 4.6-10.

The equipment and arrangement are comprised of the following:

1. CRD system pump suction piping from the Condensate and Feedwater Heating System (CFS) or the Condensate Storage Tank (CST).
2. Two 100% capacity trains. One train is normally operating, and the other train is in standby. Each train consists of the following:
  - a. One 100% capacity purge water pump with a suction and discharge filter
  - b. One 100% capacity charging water pump with a suction and discharge filter
  - c. Each purge pump has a minimum flow line at its discharge which is routed to the condenser or the CST
  - d. Each purge water and charging water pump has discharge pressure sensor that provides indication locally and in the MCR
  - e. Each charging water pump has a bypass line at its discharge which is routed to the CST
  - f. The distribution piping, branching off the main discharge line is downstream of the purge water pump discharge filter, and provides:
    - i. Purge water flow to the SDC pumps
    - ii. Purge water flow to the reactor water level reference leg instrument lines
    - iii. Sample water to the PRM system
  - g. Downstream of the purge water filters is a pressure sensor used to start the standby purge pump in the event of failure of the in-service purge pump
  - h. A flow element and sensor are used to control the position of a flow control valve which controls the total purge water flow rate to all the HCU's
3. The common purge water header directs purge water flow to the individual HCUs, where it is then directed to the FMCRDs via the scram insert piping.
4. The common charging water header includes a bladder-type nitrogen accumulator used to maintain charging water header pressure when the charging pump is not running.
5. Piping connections are provided between the individual HCUs and their respective FMCRDs
6. A scram air header, containing redundant pressure control valves, is connected to the scram valves in the individual HCUs
7. Solenoid-operated ARI valves are positioned in the scram air header to depressurize and isolate the scram air header on receipt of a scram signal

#### **4.6.2.2.3 Control Rod Drive System Operation**

The operating modes of the CRD system are described in this subsection.

##### **Normal Operation**

Normal operation is defined as operating periods where the CRD system provides charging pressure to the HCU and supplies purge water to the FMCRDs, the SDC pumps, the NBS reactor water level reference leg instrument lines, and the PRM.

One of the two purge water pumps is operating to pressurize the purge water header from the CFS or CST. The other purge water pump is shut down and on standby. A constant portion of the purge water pump discharge is continuously bypassed back to the condenser or CST in order to maintain a minimum flow through the pump. This prevents overheating of the purge pump if the discharge line is blocked. The standby purge water pump provides a full capacity backup capability to the operating pump and starts automatically if failure of the operating pump is detected by pressure instrumentation located in operating pump's discharge piping. The purge water flow control valve automatically regulates the purge water flow to the FMCRDs. The purge water flow rate is indicated in the MCR.

To maintain the ability to scram, the charging water header pressure is maintained by one of the two charging water pumps. The charging pumps are designed for intermittent operation. These pumps also draw water for the CFS or CST, with pump bypass line returning to the CST. The charging water header is maintained at the pressure required for the scram accumulators to scram the control rods. The HCU scram valves remain closed except during scram. During normal operation, no flow passes through the charging water header. A separate accumulator, located on the charging water header, assists with maintaining the header at the required pressure. When intermittent pump operation is used the pump automatically starts on low charging water pressure and stops when pressure is restored. Pressure in the charging water header is monitored continuously and displayed in the MCR along with a low-pressure alarm.

##### **Control Rod Insertion and Withdrawal**

The FMCRD design provides the capability to move a control rod in fine steps. Normal control rod movement is under the control of the RC&IS. The RC&IS controls the input of actuation power to the FMCRD motor from the electrical power supply to complete a rod motion command. The FMCRD motor rotates a ball screw, which in turn causes the vertical translation of a ball nut on the ball screw. This motion is transferred to the control rod via a hollow piston that rests on the ball nut. Thus, the hollow piston with the control rod is raised (inserted) or lowered (withdrawn) depending on the direction of rotation of the FMCRD motor and ball screw.

During a control rod insertion, the solenoid-operated purge water makeup valve within the associated HCU is opened to increase the purge water flow to the FMCRD. The increased flow offsets the volumetric displacement within the FMCRD as the hollow piston is inserted into the core and prevents reactor water from being drawn back into the drive.

##### **Scram**

In response to an automatic or a manual scram the power is interrupted to both scram solenoid valve assembly coils which vent the air from the scram valve operator in each HCU. Venting of the air causes the scram valves to open by spring action. When the scram valve opens in the associated HCU, hydraulic pressure provides an insertion force to the respective FMCRD. When the hydraulic force is applied, the FMCRD's hollow piston lifts off the ball nut and inserts the control rod rapidly. The water displaced from the FMCRD is discharged into the RPV. Indication that the scram has been successfully completed (i.e., all rods full-in position) is displayed in the MCR.



Simultaneously with the hydraulic scram, each FMCRD motor is signaled to start from the ERICP. This signal causes the FMCRD motor to drive the ball nut upward to a position just below the fully inserted and latched hollow piston. This action is known as the scram follow function.

After reset of the scram logic, each scram valve recloses and the CRD charging water pump recharges the HCU accumulators.

Reactor scram logic is described in Chapter 7.

### **Select Control Rod Rapid Insertion**

During SCRRI, a predefined set of control rods insert rapidly. Upon receipt of the SCRRI initiation signal, the CRD purge water flow control valve opens to provide increased purge water flow to the associated HCUs. At those HCUs, the solenoid-operated purge water makeup valve opens to increase the purge water flow to the FMCRD to prevent reactor water from being drawn back into the drives.

### **Alternate Rod Insertion**

The ARI function of the CRD System is initiated by the ERICP which provides an alternate, diverse and independent means for actuating hydraulic rod insertion. The signals to initiate the ARI function are the same as those for automatic and manual scrams. Following receipt of any scram signal, the ARI solenoid-operated valves on the scram air header actuate to depressurize the scram air header which opens all HCU scram valves. The FMCRDs then insert the control rods hydraulically. The same signals that initiate ARI simultaneously initiate an FMCRD scram follow function. Each vent path has two valves in series such that both valves are required to actuate to cause depressurization to prevent spurious ARI actuation.

#### **4.6.2.3 Instrumentation and Control**

##### ***4.6.2.3.1 Safety Class 1 Instrumentation and Control System***

The SC1 I&C system is described in Chapter 7.

##### ***4.6.2.3.2 Safety Class 2 and 3 Instrumentation and Control System***

The SC2 and SC3 instrumentation and control for the CRD system is described in Chapter 7, Section 7.3.

#### **4.6.2.4 Power Supplies**

The SC2 and SC3 electrical distribution system provides power to the CRD purge pump motors, CRD charging water pump motors, and CRD FMCRD motors.

#### **4.6.2.5 Qualification**

Equipment qualification requirements, including environmental, dynamic, functional, and Electromagnetic Compatibility (EMC) is described in Chapter 3, Section 3.9. The qualification requirements are based on equipment classifications described in Chapter 3, Section 3.2, Appendix 3A.

### **4.6.3 Safety Evaluation of the CRD System**

#### **4.6.3.1 Scram Time**

The control rod scram function of the CRD system provides the negative reactivity insertion required by the Safety Design Bases in Subsection 4.6.1. The required scram time is used in the Chapter 15 safety analyses.

#### **4.6.3.2 Scram Reliability**

Key system features resulting in high scram reliability include:

- Scram valves open by spring action and are normally held closed by pressurized control air
- To cause hydraulic scram, a de-energizing reactor trip signal is provided to solenoid-operated pilot valves that vent the control air from the scram valves for opening
- The SC1 I&C hydraulic scram is designed so that the HCU scram signal independently initiates a hydraulic scram demand, from whatever source, regardless of any other rod positioning signal
- The FMCRD hollow piston and guide tube are designed so they do not restrain or prevent control rod insertion during scram
- Each FMCRD mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram upon receipt of I&C signal. This provides a diverse means to assure control rod insertion
- The system is “fail-safe” in that loss of either electrical power to the scram solenoids, or loss of control air pressure to the scram valve operator, causes a scram
- Departure from the ball-nut releases spring-loaded latches in the hollow piston to engage slots in the guide tube
  - These latches support the hollow piston in the fully inserted position
  - Following a hydraulic scram insertion, the control rod cannot be withdrawn until the ball-nut is driven up, re-engaged, and the hollow piston de-latched from the guide tube
- The design also includes ARI pilot valves on the control air header, which serves all 29 scram valves
  - The ARI pilot valves are energized-to-actuate, and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods

#### **4.6.4 Testing, Inspection, and Maintenance**

The BWRX-300 inspection requirements provide mandatory Preservice Inspection, In-Service Inspection (ISI), and In-Service Testing (IST) requirements for components and systems. Preservice Inspection, ISI, and IST requirements are consistent with ASME BPVC, Section XI and ASME Operation and Maintenance of Nuclear Power Plants (OM Code).

The Preservice Inspection, ISI, and IST requirements include examinations, inspections, and testing of the SC1 portions of the CRD system that are designed and installed in accordance with ASME BPVC Section III, applicable Canadian codes and standards, and International Atomic Energy Agency (IAEA) Safety Standards. In addition, this specification includes basic design control considerations within ASME NQA-1, Part 2.

Any required surveillance and post-maintenance testing are determined as part of the work planning / work release approval process.

#### **FMCRD Maintenance**

The FMCRD is designed to permit coupling and uncoupling of the control rod from either below the reactor vessel without necessitating the removal of the reactor vessel head or above the reactor vessel with the reactor vessel head removed.

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

Maintenance procedures prohibit coincident removal of the control rod and FMCRD of the same assembly. In addition, contingency procedures address core and spent fuel cooling capability and mitigative actions during FMCRD replacement with fuel in the vessel.

The FMCRD design also allows for separate removal of the motor unit, position indicator probe, Separation Indicator Probe, and lower component for maintenance during plant outages without disturbing the upper assembly of the drive. While these FMCRD components are removed for servicing, the associated control rod is maintained in the fully inserted position by one of two mechanical locking devices that prevent rotation of the ball screw and drive shaft.

#### **4.6.5 References**

None.

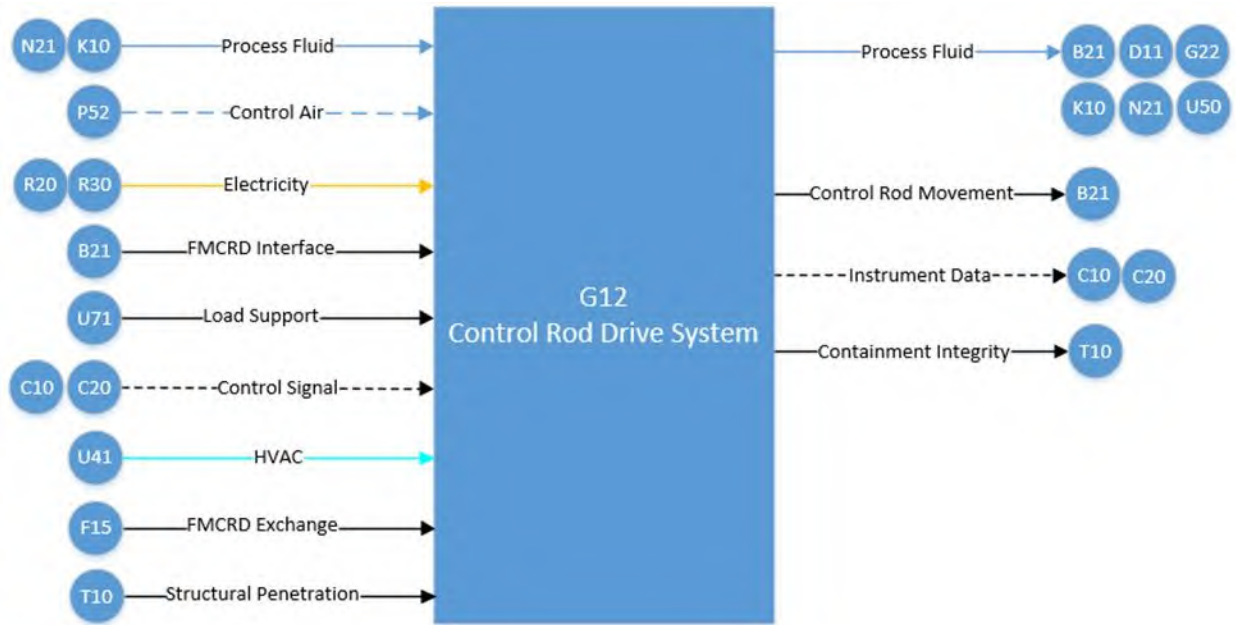
NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 4.6-2: Control Rod Insertion Versus Time**

Reactor Vessel Bottom Gauge Pressure MPaG (psig)	Rod Insertion Position	Required Maximum Time (s) <sup>(2)</sup>
≤ 7.48 MPaG (1085 psig)	10 %	≤ 0.42
	40 %	≤ 1.00
	60 %	≤ 1.44
	100 %	≤ 2.80
≤ 8.75 MPaG (1269 psig)	10 %	≤ 0.46
	40 %	≤ 1.20
	60 %	≤ 1.71
	100 %	≤ 3.70
≤ 9.48 MPaG (1375 psig)	10 %	≤ 0.56
	40 %	≤ 1.40
	60 %	≤ 2.03
	100 %	≤ 4.20

(2) All times are after deenergizing the scram solenoids

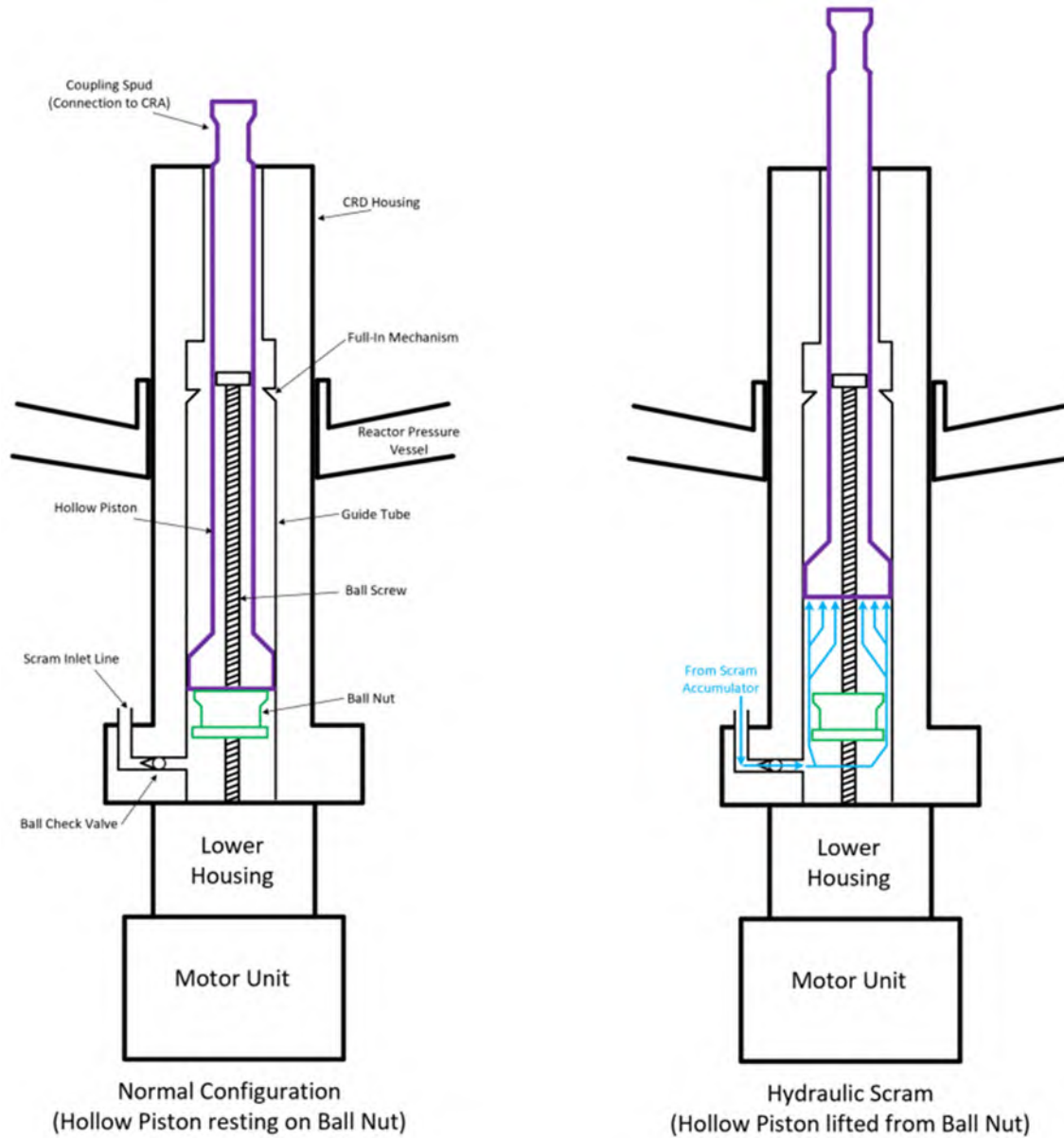
NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION



**Legend:**

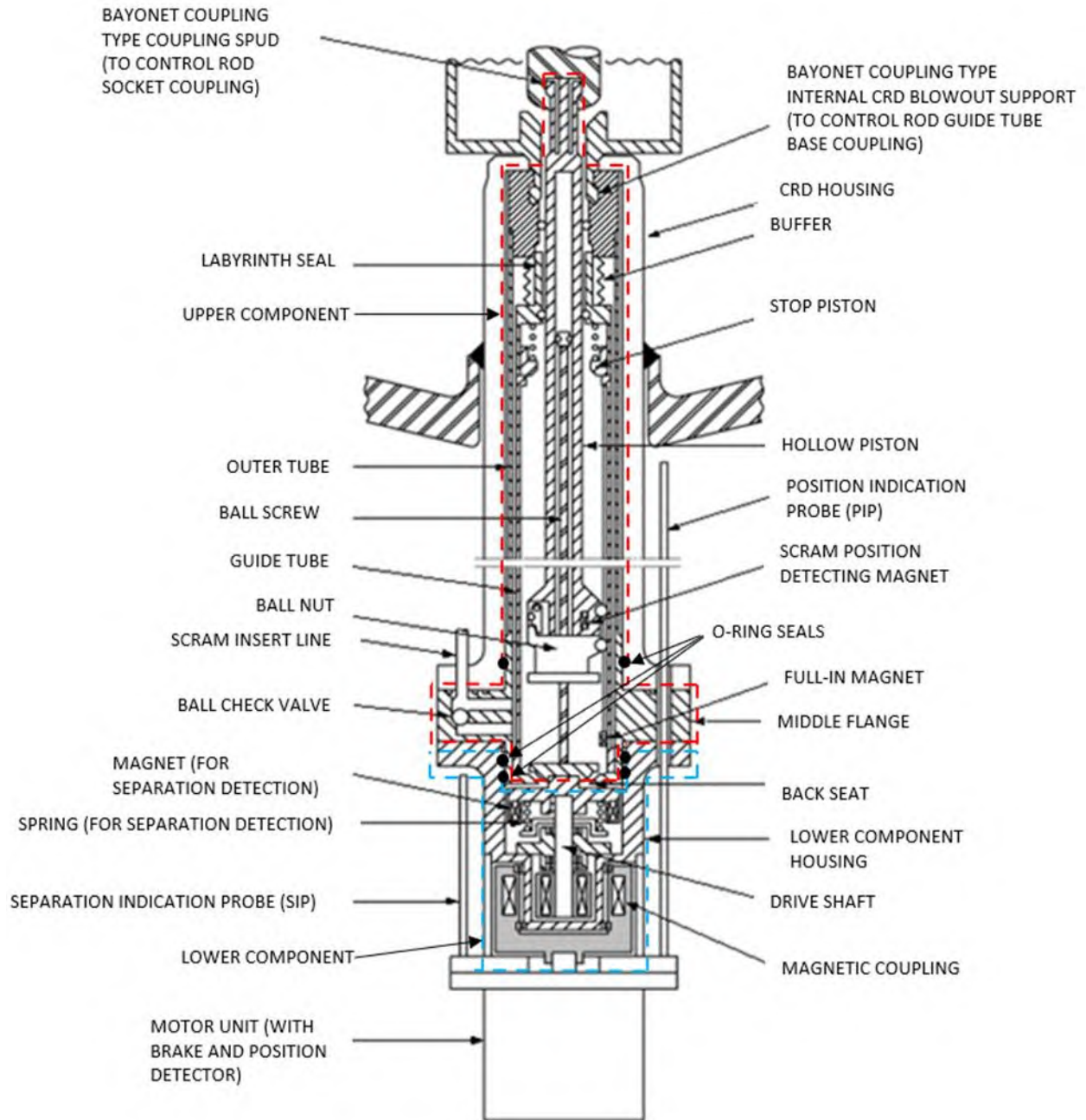
B21 Nuclear Boiler System  
C10/C20 Safety Class 1/2 Instrumentation and Control System  
D11 Process and Radiation Monitoring  
F15 Refueling Equipment and Servicing  
N21 Condensate and Feedwater System  
K10 Liquid Waste Management System  
P52 Plant Pneumatics System  
R20 Safety Class 2 and 3 Electrical Distribution System  
R30 Non-Safety Electrical Distribution System  
T10 Primary Containment  
U71 Reactor Building Structure  
U41 Heating Ventilation and Cooling System  
U50 Equipment and Floor Drain System

**Figure 4.6-1: CRD Interfaces Diagram**

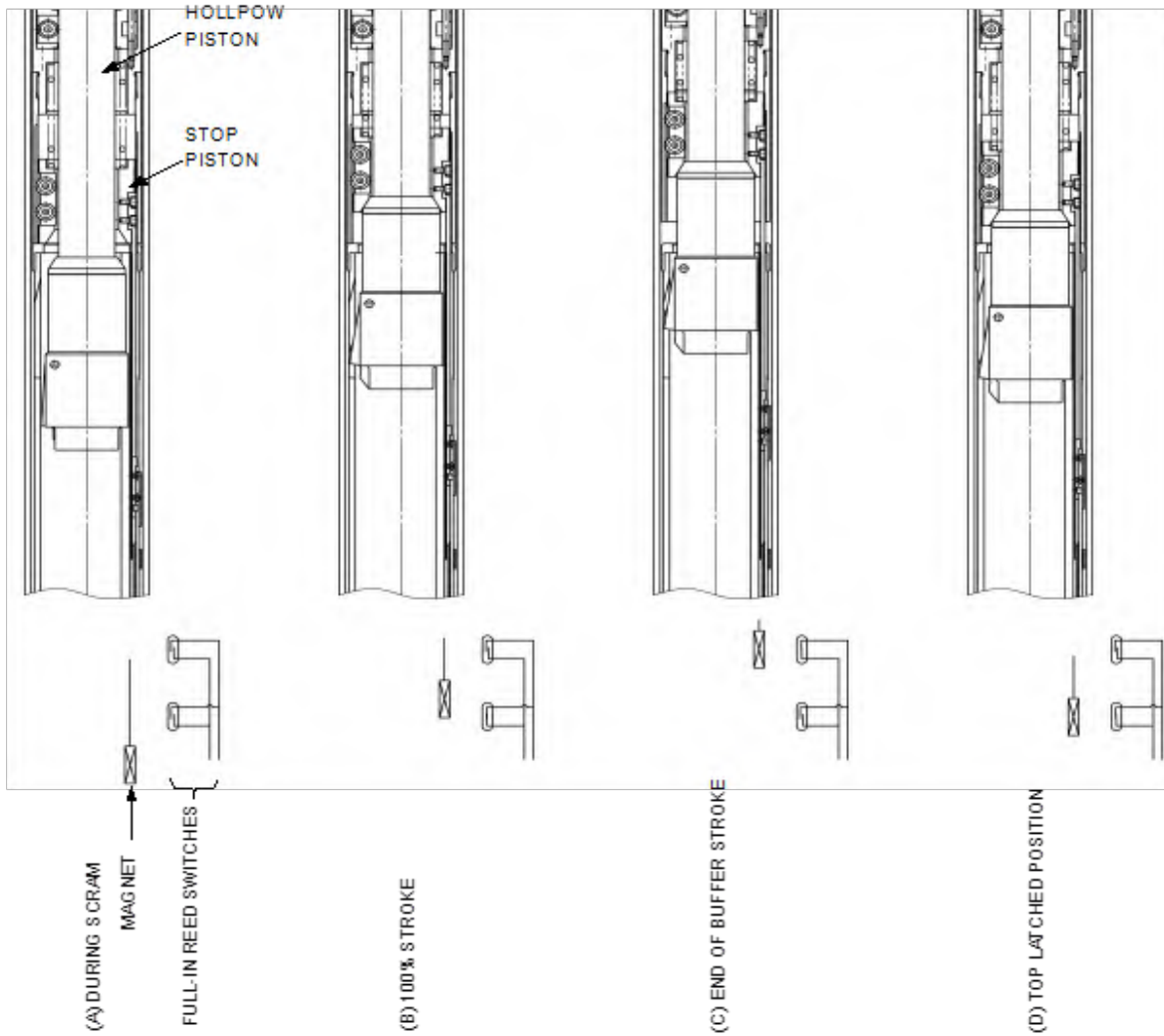


**Figure 4.6-2: Magnet Coupling Fine Motion Control Rod Drive Cross-Section**

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 4.6-3: Schematic Representation of Magnet Coupling Fine Motion Control Rod Drive**



**Figure 4.6-4: Representative Continuous Full-in Indicating Device**



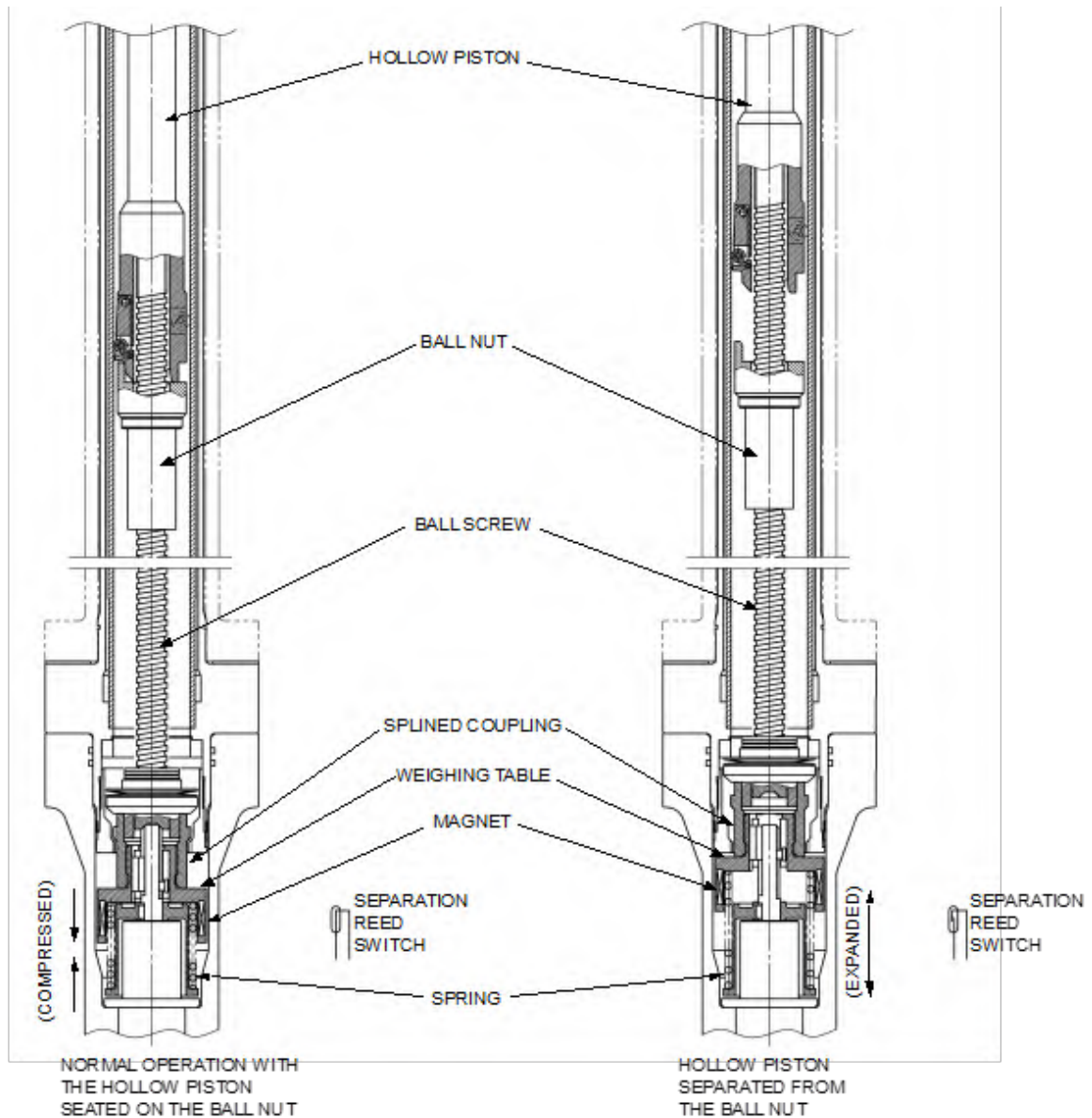
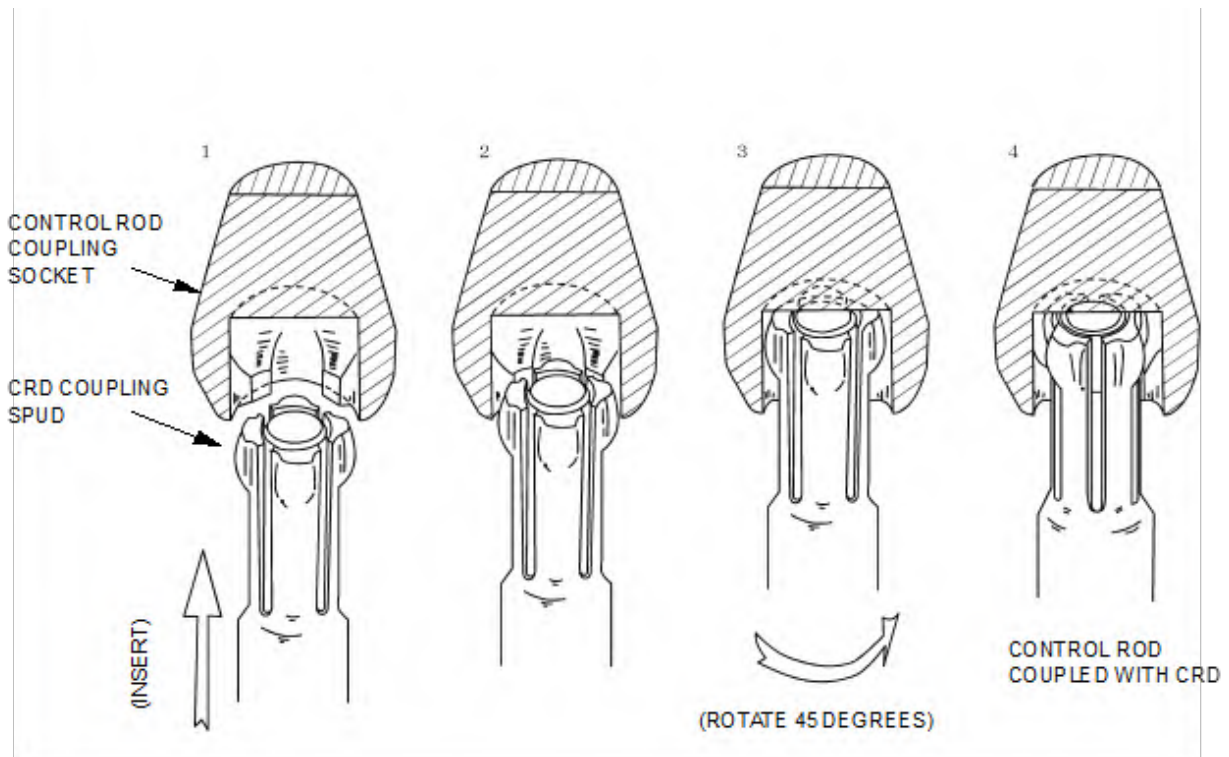
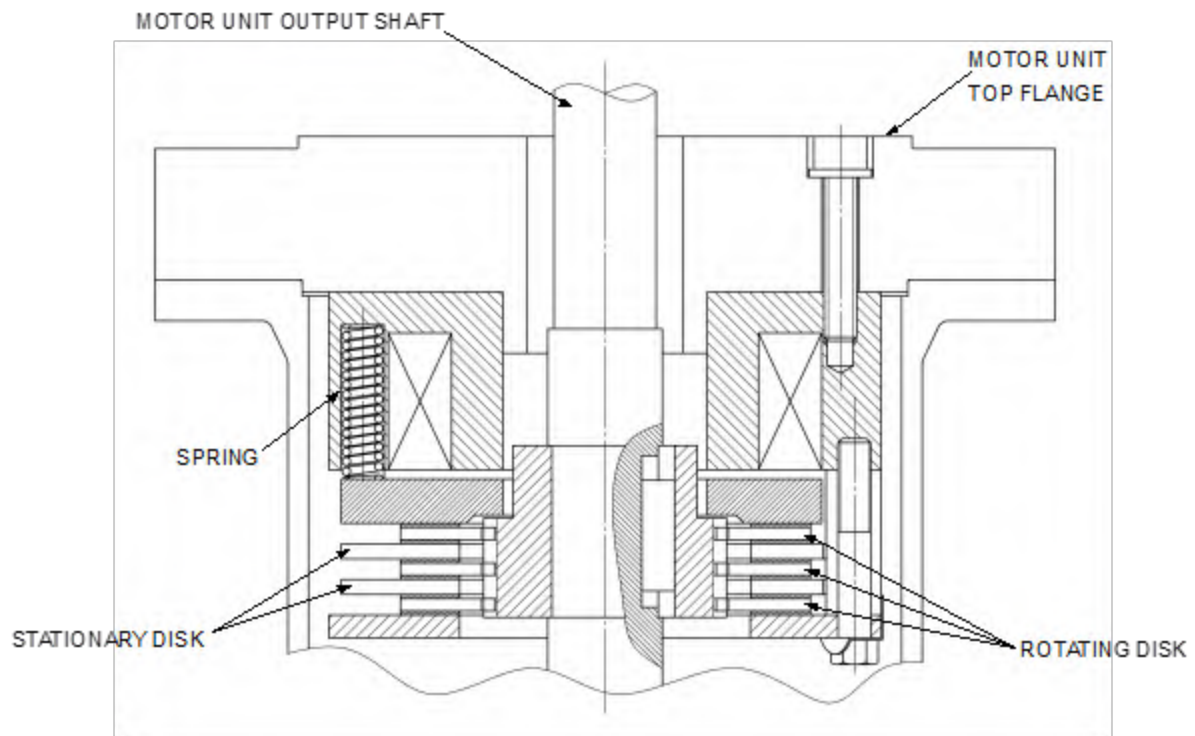


Figure 4.6-5: Representative Control Rod Separation Detection



**Figure 4.6-6: Representative Control Rod to Control Rod Drive Coupling**



**Figure 4.6-7: Representative FMCRD Electro-Mechanical Brake**

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

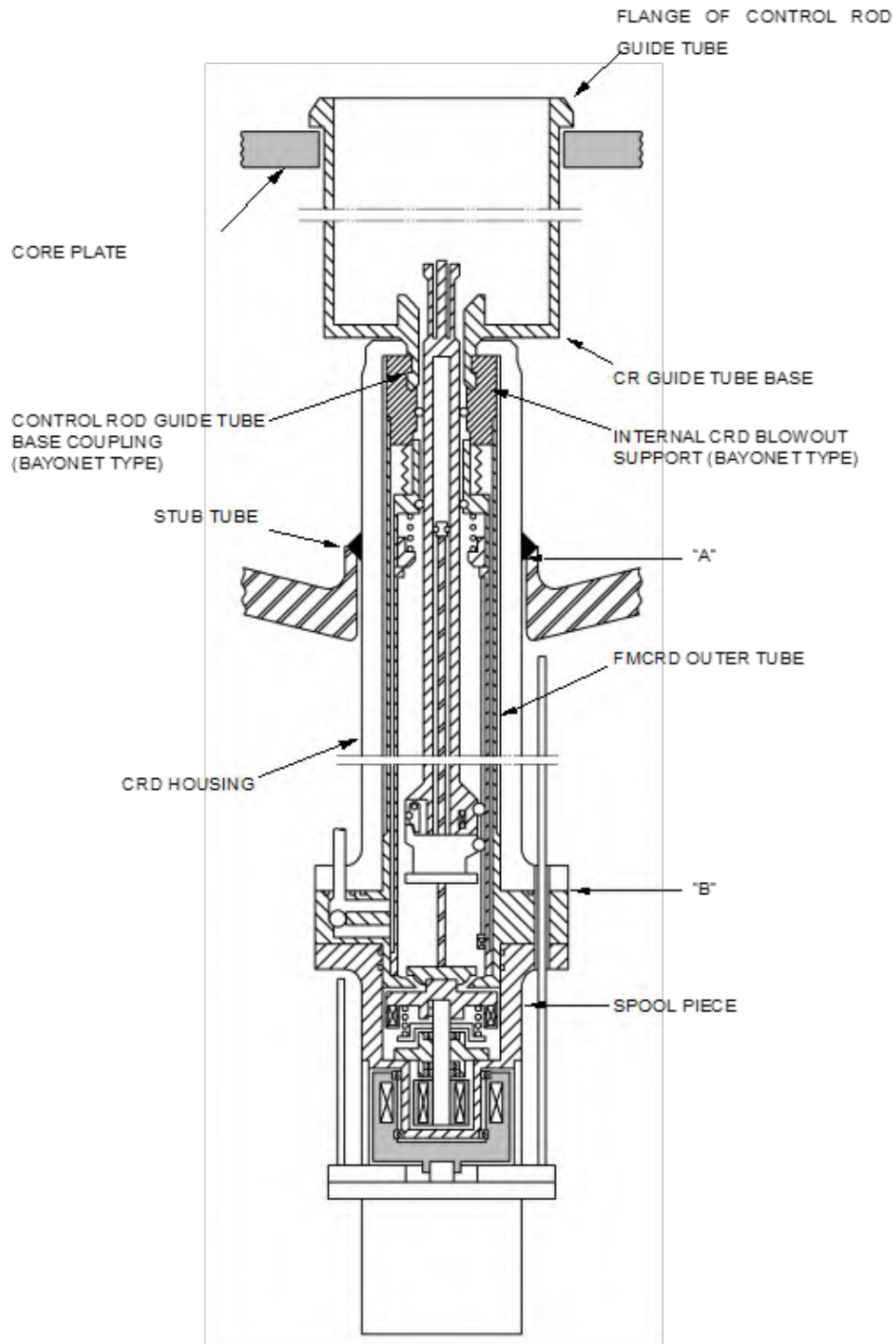


Figure 4.6-8: Representative Internal CRD Blowout Support Schematic

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

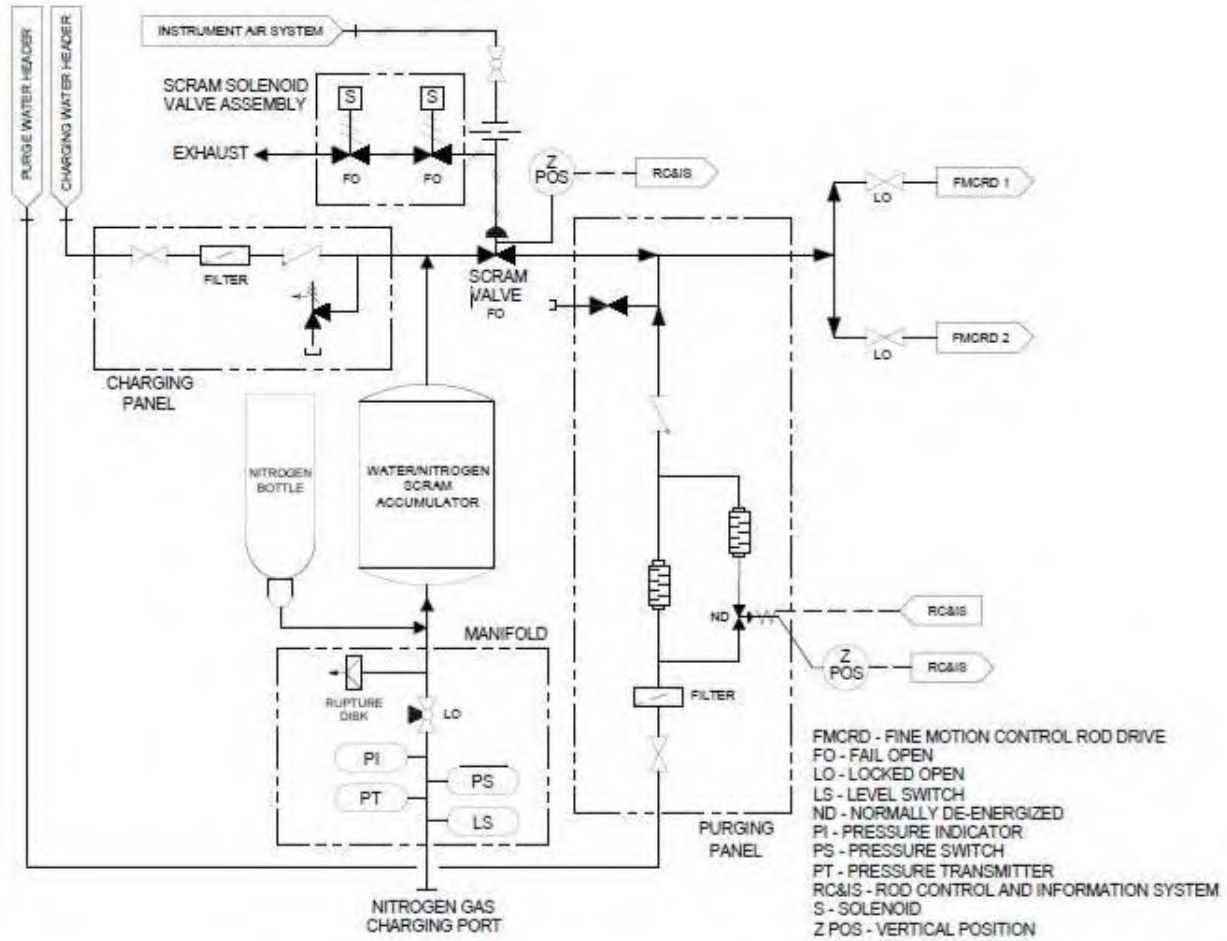


Figure 4.6-9: HCU Simplified P&ID

NEDO-33953 REVISION 0  
NON-PROPRIETARY INFORMATION

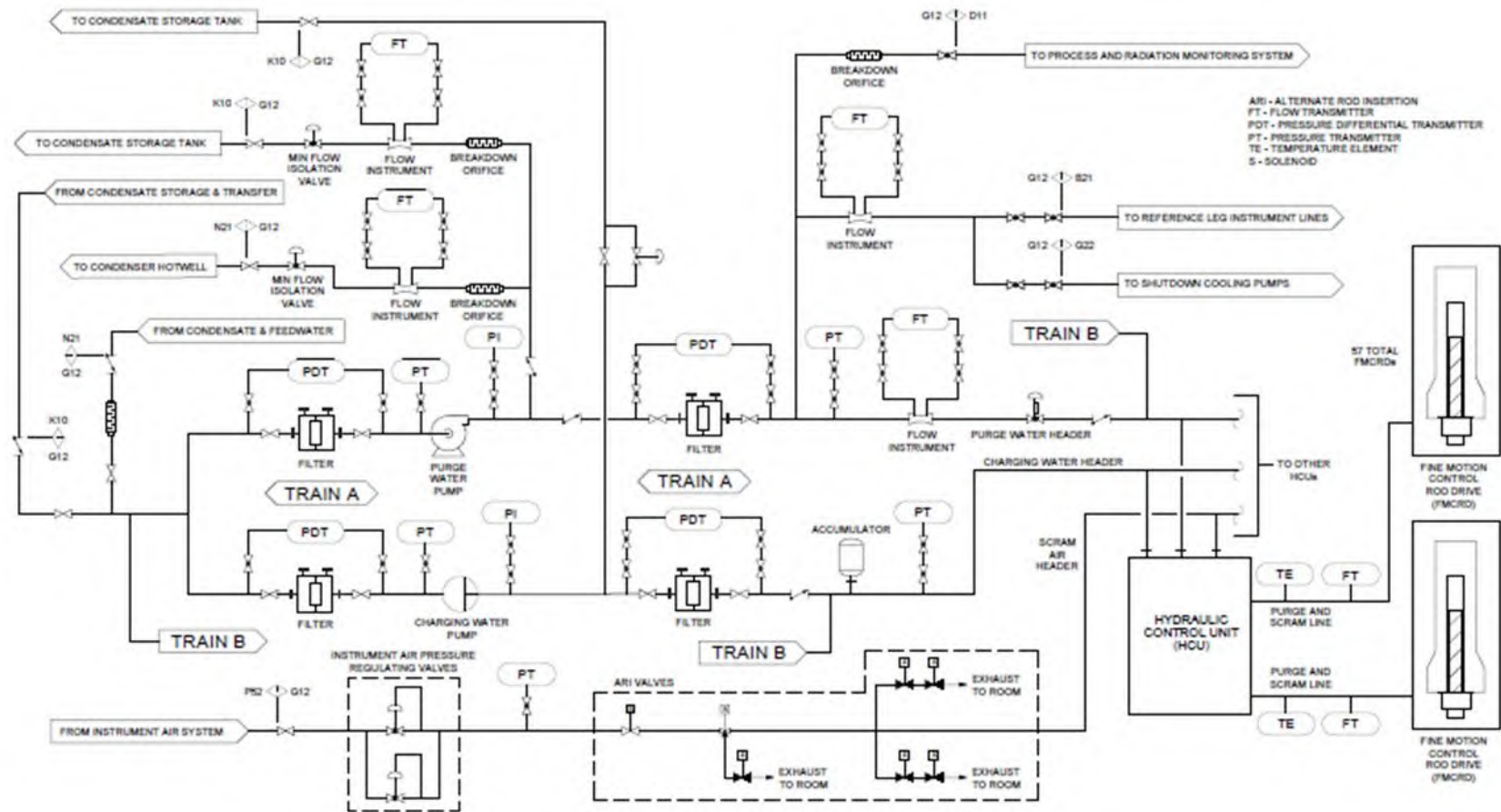


Figure 4.6-10: CRD Simplified P&ID



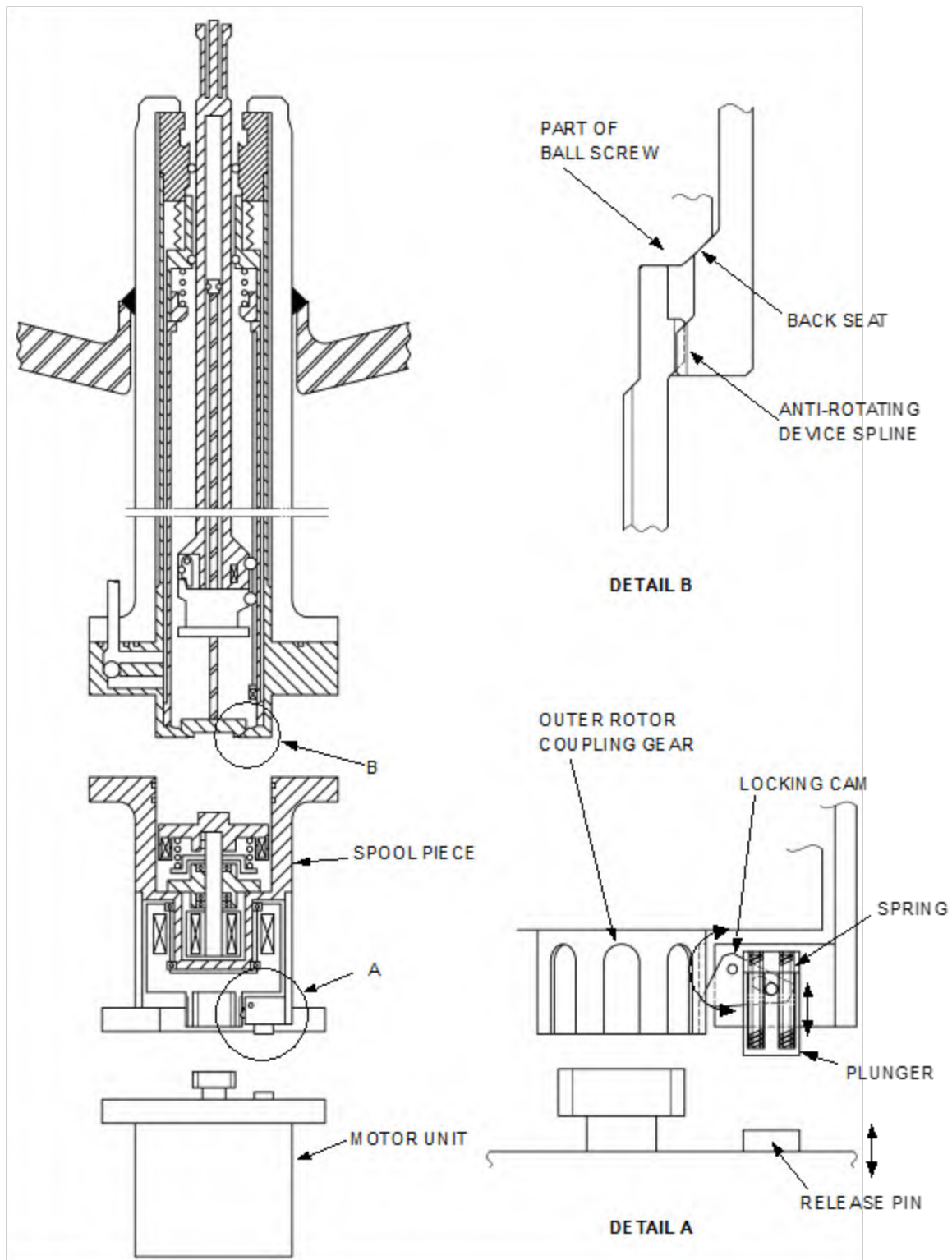


Figure 4.6-11: Representative FMCRD Anti-Rotation Devices

## **4.7 Core Monitoring**

Core monitoring is a function of the plant computer system that provides three-dimensional core power monitoring. Core monitoring provides confidence that the plant is operating in conformance to SAFDL. Core monitoring obtains instrumentation information from the Distributed Control and Information System (refer to Chapter 7, Subsection 7.3.3.2), calculates power distributions and resulting thermal limits. These power distributions are adapted to signals from plant GTs and LPRMs as applicable.

### **4.7.1 System and Equipment Functions**

The core monitoring function acquires live reactor data from site plant data acquisition systems as required, to define the reactor state for use by the core simulator. The acquired data are validated within acceptable ranges. Core monitoring calculates the accumulated thermal and electrical energy produced by the plant from the beginning of an operating cycle. On a periodic basis, upon user request, or when triggered by a system event, a core simulator determines current core characteristics based on the current reactor state and the previous history of the following core components:

- Thermal limit monitoring
- Fuel conditioning
- GT processing
- Prediction at achievable operating regimes
- Tracking of local and global xenon behavior
- Identification of LPRM drift
- Power adaption based on in-core power measurements
- Graphical display of data
- Power / flow tracking
- BWR operating guidelines implementation
- Isotopic tracking

Core parameters are calculated either by the core simulator or during post processing (including parameters associated with fuel bundles, fuel channels, GTs, LPRMs, and control rods). The core monitoring function then compares these core parameters against technical specifications, licensing limitations, manufacturer's guidelines, and site-specific limits. If user actions are required, the system generates alerts, warnings, and notifications. The primary core monitoring functions are:

- Calculation of current core parameters
- Prediction of future core parameters
- Calculation of LPRM calibration factors
- Calculation and tracking of the isotopic inventory of the fuel
- Generation of visualizations and reports on core performance
- Generates core technical information for use by other plant systems as required



The core monitoring function uses live reactor and in-core information, and coupled with a three-dimensional BWR simulator model, produces current and predicted core performance information. When the live plant input data, including LPRM, control rod position, and core power are not available, the core monitoring function still performs a core performance calculation using the Manual Monitor option. The performance information includes standard visualizations such as thermal-hydraulic parameters, thermal margins, fuel conditioning margins, fuel burnup, LPRM information, control rod position information and power/flow maps. Information from the core monitoring function is used by the Licenced Plant Operator and the Reactor Engineer in the reactor operational decision-making process, for both steady-state operation and during plant maneuvers, to ensure compliance with licenced limits.

Core monitoring also supports the calibration process for LPRMs by processing GT data and live LPRM data to yield individual LPRM calibration factors. The core monitoring function does not directly calibrate the LPRMs, instead the calibration adjustments require operator action to be implemented in the LPRM systems.

The core monitoring function uses a coupled nuclear thermal-hydraulic diffusion theory model, to provide a three-dimensional simulation of BWR core characteristics performance. The accuracy of the modeling is enhanced by adaptive algorithms that conform results to measured plant data from the LPRMs and GTs. Instrument signals are compared to simulator predictions and outlier data are rejected as anomalous. Unavailability or failure of a limited number of LPRM and GT signals does not significantly affect plant operation.

#### **4.7.1.1 Safety Design Bases**

The core monitoring function provides a calculation of core performance parameters, such as the Maximum Fraction Limiting Critical Power Ratio (MFLCPR), the Maximum Fraction of Linear Power Density (MFLPD), and margin to fuel conditioning limits. These values are monitored and used by Reactor Engineers and Licenced Plant Operators to ensure that, during steady-state operation and any reactor maneuvers, these parameters are within the licenced and administrative plant limits.

#### **4.7.1.2 Licencing Basis and Operational Experience**

The current core monitoring function embodiment for the operating fleet and the BWRX-300 is named ACUMEN [GESTAR II Rev 23 with ACUMEN SLMCPR AM 42 approval by the U.S. NRC]. ACUMEN has been performing as the official core monitoring function of record at three BWRs since March and April of 2017 and October of 2019, respectively. In all these applications, ACUMEN replaced 3D Monicore and has been built on the experience gained from GNF's legacy core monitoring system which has a rich history and experience of operating at 20+ BWRs for more than 30 years.

The ACUMEN core monitoring and prediction system monitors and predicts fuel performance based on neutron instrumentation inputs and provides reactor engineers and operators with data to optimize the performance of the core. ACUMEN is designed to utilize GNF's 3-D core simulator PANAC11 to compute thermal limits, predict reactor operations using ACUMEN's Predictor feature, utilize failed fuel prevention techniques in Predictor, and monitor for streamlined failed fuel management.

#### **4.7.1.3 Equipment Interfaces**

The standard core monitoring function interface provides a means by which to integrate with other systems, including, but not limited to, the plant Distributed Control and Information System. Core monitoring includes a high performance and redundant hardware platform designed to meet the requirements of high availability and redundancy, as well as implement the cyber security requirements of CNSC REGDOC- 2.5.2 (Reference 4.1-1).

#### **4.7.1.4 Core Monitoring Operation**

Plant operating conditions that utilize the core monitoring function includes reactor shutdown, ascension to power, steady-state operation, and reactor maneuvering.

In the reactor shutdown mode, all control rods are inserted, the reactor fission power is essentially zero, and the core is subcritical. In this condition, the core monitoring function can perform predictive calculations to confirm required shutdown margin.

At initial start-up of the reactor, the core monitoring function performs predictive calculations of core reactivity for each incremental control rod withdrawal during the approach to initial criticality. High incremental changes in reactivity associated with steps in the control rod withdrawals that would cause a short reactor period can be identified and mitigated. The core monitoring function calculations address the impact of the actual coolant temperature associated with the criticality calculations.

Once the reactor power reaches approximately 5% of the rated thermal power level, the live plant thermal-hydraulic data (temperatures, flow rates, pressure) for the reactor system and the LPRM data become available. The details of the live data and the plant heat balance is monitored by the core monitoring function.

Once the heat balance is stable, the core monitoring function can monitor the core power distribution and margin to operating guidelines and thermal limits (MFLCPR, MFLPD). In addition to providing predictions of the core power distribution, the core monitoring function also tracks the nodal xenon concentration in the core. It is generally not required to monitor thermal margins (e.g., MFLCPR) at very low power levels.

Once the neutronic instrumentation data are judged validated, core monitoring performs a power distribution calculation and adapts the core power distribution calculation to the data. LPRM readings are compared to the GT data, and LPRMs are calibrated to match the full-core normalized GT data.

During power ascension, control rod adjustments are made. The core monitoring function is used during the power ascension to compute core performance so that thermal margins and fuel conditioning limits can be monitored. The Predictor function of core monitoring can be used to analyze and optimize the planned power ascension. The core monitoring and Predictor functions used during plant start-up are also used during plant maneuvers, including load following, periodic maintenance and/or periodic control rod sequence maneuvers during the operating cycle.

#### **4.7.2 References**

None.

## **4.8 Thermal-Hydraulic Stability**

### **4.8.1 Background**

Under certain conditions, BWRs can be susceptible to coupled neutronic/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). If the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

### **4.8.2 Design Bases**

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that SAFDL are not exceeded during any condition of normal operation, including the effects of AOOs.

The reactor core and associated coolant, control, and protection systems are designed to assure that power oscillations that could result in conditions exceeding SAFDL are either not possible or can be reliably and readily detected and suppressed.

### **4.8.3 Types of BWR Oscillations**

There are two types of oscillations associated with BWR stability.

Type 1 instabilities experienced during startup do not result in a reactivity/power response. Type 1 oscillations are characterized by initiation of vapor production in the chimney region leading to a reduction in hydrostatic head in the chimney and a resultant core flow increase, which, in turn, could cause voids to collapse in the chimney. The BWRX-300 reactor goes through an unstable phase during startup. This type of oscillation is unavoidable in a natural circulation reactor because the unstable power/flow region must be crossed prior to establishing a steady two-phase voided region in the chimney; however, the magnitude of the flow oscillations is typically very small. As Type 1 oscillations do not result in a change in core moderator density, there is no power response and, as a result, the Fuel Cladding Integrity Safety Limit is maintained.

Type 2 oscillations are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). In Type 2 instability, if the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

Within the Type 2 domain, there are three (3) modes of oscillation:

1. Core-wide (or in-phase) mode oscillations are dominated by the fundamental mode of the neutron dynamics and are detected by the Average Power Range Monitor (APRM) system.
2. Regional (or out-of-phase) mode oscillations, with parallel channel momentum dynamics as the dominant feedback loop and reinforced by a subcritical mode of the neutronics, are not easily detected by the APRM system since the reactor powers in each half of the reactor core along the harmonic axis of oscillations tend to cancel out each other.
3. Single channel thermal-hydraulic mode oscillations are related to the momentum dynamics of heated channel in a two-phase flow regime. The oscillations are purely hydraulic in nature, and with no power response, they are not easily detected by the APRM system.

#### 4.8.4 Design Criteria

The most limiting stability condition in the BWRX-300 normal operating region is at the rated power/flow condition. The BWRX-300 is designed so that the core remains stable throughout the entire operating domain. In the time domain analysis, decay ratio is defined as the ratio of the amplitude of the first two successive peaks. For the BWRX-300, the decay ratio is averaged over the first few peaks.

Conservative design criteria are imposed on the core-wide, regional, and single-channel decay ratios under all conditions of normal operation and anticipated transients. The limiting mode (i.e., highest decay ratio) for thermal-hydraulic instabilities (Type 2 oscillations) is the core-wide mode. The channel and regional modes are highly damped.

#### 4.8.5 Stability Solution Design

The stability licensing criterion for all nuclear power plants is set forth in CNSC REGDOC-2.5.2, Section 8.1 (Reference 4.1-1) requiring assurance that the safety functions prevent unacceptable instabilities during normal operation, AOO conditions, and DBA conditions. An unacceptable oscillation is characterized by an amplitude that grows and may challenge SAFDL. CNSC REGDOC-2.5.2 (Reference 4.1-1) also stipulates that power oscillations that can result in conditions exceeding SAFDL are reliably and readily detected and suppressed. The limiting stability condition in the BWRX-300 during normal power operation region is at the rated power. Therefore, the BWRX-300 is designed so that unacceptable coupled neutronic and thermal-hydraulic power oscillations are not possible throughout the entire operating region. The BWRX-300 is designed to protect acceptable fuel design limits during AOOs. As a Defence-in-Depth feature, the BWRX-300 implements a stability monitoring function to detect instabilities should they occur (Refer to Chapter 7, Subsection 7.3.3) to inform operator action. This solution is similar to the Option I-D long-term stability solution in the U.S. (Reference 4.4-8).

The BWRX-300 design has features that result in stable behavior in normal operation and minimize the effects of potential oscillations in off-normal conditions include:

- Small Core

The small core size and higher inlet orifice pressure drop of the BWRX-300 reduces the likelihood of regional mode instabilities. Conservative analyses are performed to confirm that regional mode oscillations are not possible and that core oscillations would be core-wide dominant. Any unacceptable core-wide oscillation is mitigated by the high flux scram.

  - Tighter neutronic coupling precludes regional mode oscillations
  - Core-wide oscillations are the dominant mode
- Natural circulation
  - No recirculation pump trips that result in significant change from stable to unstable conditions
  - Loss of Feedwater Heating (LFWH) AOO impact on stability is mitigated by SCRR operation for a feedwater temperature reduction of 16.7°C or higher. The SCRR function is described in Chapter 7, Subsection 7.3.3.2. The loss of feedwater heating AOO analysis is described in Chapter 15, Subsection 15.5.3.1.1

- Tall chimney
  - Increases volume of water
  - Increases driving head and natural circulation flow
  - Dampens oscillations
- Large downcomer area
  - Reduces flow restrictions
- High inlet orifice pressure drop
  - Improves two-phase to single phase pressure drop ratio
- Balanced feedwater temperature
  - Neither thermal margins nor decay ratios are compromised
  - Minimizes normal operation inlet subcooling
- Less subcooling

Compliance with CNSC REGDOC-2.5.2 (Reference 4.1-1) is assured by implementing design criteria for the decay ratio. The stability acceptance criterion for the current GEH fleet of BWRs is a calculated core decay ratio of  $\leq 0.80$  for normal operation and AOOs considering uncertainties. The BWRX-300 has been established to provide margin to the decay ratio criterion.

#### **4.8.6 Stability Analysis Methods**

A detailed discussion of the methods used to analyze BWRX-300 thermal-hydraulic stability is presented in TRACG Application for BWRX-300 (Reference 4.8-1).

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG uses a multi-dimensional, two-fluid model for the reactor thermal-hydraulics and a three-dimensional reactor kinetics model. The models can be used to accurately simulate a large variety of test and reactor configurations. These features allow for realistic simulation of a wide range of BWR phenomena and are described in detail in the TRACG Model Description Licensing Topical Report (Reference 4.3-3).

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and operating BWR plant data. The details are presented in the TRACG Qualification Licensing Topical Report (Reference 4.4-7). The USNRC has approved TRACG for application to Economic Simplified Boiling Water Reactor (ESBWR) stability analysis, NEDC-33083P-A (Reference 4.4-6).

TRACG is qualified to accurately model natural circulation for a wide application range that encompasses the BWRX-300. The TRACG qualification bases, NEDE-32177P (Reference 4.4-7) include benchmarking natural circulation flow and instability onset (FRIGG-4 FT-36C Onset of Instability Tests) in which TRACG conservatively predicts the power at the onset of limit cycle oscillations (over a range of system pressures and inlet subcooling) to be reasonably close to measured values.

#### **4.8.7 Stability Evaluation**

To determine whether core-wide oscillation is the dominant mode, the flow velocity for symmetric channels on opposite sides of the core are perturbed out of phase. If the core is not susceptible to regional mode oscillations after such a flow velocity perturbation, then symmetric out of phase channels come into phase after a short duration. This analysis confirms that regional mode oscillations are not possible. As such, only the core-wide mode is applicable to the BWRX-300.

##### **4.8.7.1 Stability Performance During NO**

Stability analyses are performed at rated conditions and at multiple exposure points during the cycle. Stability analyses are performed by perturbing the system and assessing the core response. Such perturbations might include pressure pulse perturbation or variation in feedwater temperature. The response to a pressure perturbation in the steam line is analyzed to obtain the decay ratio for the BWRX-300 stability analysis.

The resulting decay ratios affirm stable operation, with margin to the stability decay ratio criteria, as described in Chapter 15, Subsection 15.5.2.4.

##### **4.8.7.2 Stability Performance Evaluation During AOOs**

In general, the stability margin reduces when the reactor power to flow ratio increases and/or core flow reduces or the core inlet subcooling increases (that also results in power increase). As the BWRX-300 is a natural circulation reactor, recirculation pump trip transients are not applicable and the key state variable that affects decay ratio is the inlet subcooling. As such, the loss of feedwater heating AOO is the limiting transient for stability.

The loss of feedwater heating event increases core inlet subcooling (i.e., the inlet temperature decreases) and increases core power. The BWRX-300 has been established to initiate a SCRR in response to a loss of feedwater heating AOO that mitigates the increase in core thermal power. Stability analyses for the LFWH AOO with SCRR affirm the stability decay ratio criterion is met and the results are summarized in Chapter 15, Subsection 15.5.2.4.

#### **4.8.8 References**

- 4.8-1 NEDC-33987P, "BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application," GE-Hitachi Nuclear Energy Americas, LLC.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33954

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analyses Report:**

**Chapter 5  
Nuclear Boiler System and Associated Systems**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.



NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CFS	Condensate and Feedwater Heating System
CIV	Containment Isolation Valve
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CSA	CSA Group
CUW	Reactor Water Cleanup System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL	Defense Line
DPS	Diverse Protection System
ECCS	Emergency Core Cooling System
EDM	Electro-Discharge Machining
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
FAC	Flow Accelerated Corrosion
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FW	Feedwater
FWH	Feedwater Heater
FWRIV	Feedwater Reactor Isolation Valve

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
GEH	GE Hitachi Nuclear Energy
GT	Gamma Thermometer
HAZ	Heat Affected Zone
HCU	Hydraulic Control Unit
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
ICRRIV	Isolation Condenser Return Reactor Isolation Valve
ICSRIV	Isolation Condenser Supply Reactor Isolation Valve
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
IST	In-Service Testing
LLRT	Local Leak Rate Testing
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor
LTC	Licence to Construct
LTR	Licensing Topical Report
LWM	Liquid Waste Management System
LWR	Light Water Reactor
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
NDE	Non-Destructive Examination
NPS	Nominal Pipe Size
NR	Narrow Range
NRC	U.S. Nuclear Regulatory Commission

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
OD	Outside Diameter
OLC	Operational Limits and Conditions
OLNC	On-Line NobleChem™
OPEX	Operating Experience
P&ID	Piping and Instrumentation Diagram
PAM	Post-Accident Monitoring
PAS	Plant Automation System
PCS	Primary Containment System
PFD	Process Flow Diagram
PRNM	Power Range Neutron Monitoring System
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
RTS	Reactor Trip System
SC	Safety Class
SCC	Stress Corrosion Cracking
SDC	Shutdown Cooling System
SIR	Seismic Interface Restraint
SSC	Structures, Systems, and Components
SCCV	Steel-Plate Composite Containment Vessel
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam Subsystem
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission
UT	Ultrasonic Testing
WR	Wide Range

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

Acronym	Explanation
WRNM	Wide Range Neutron Monitor

## TABLE OF CONTENTS

<b>5.0</b>	<b>NUCLEAR BOILER SYSTEM AND ASSOCIATED SYSTEMS.....</b>	<b>5-1</b>
5.1	Summary Description .....	5-1
5.1.1	Reactor Coolant Pressure Boundary .....	5-2
5.1.2	Reactor Pressure Vessel and Appurtenances.....	5-2
5.1.3	Main Steam Subsystem .....	5-3
5.1.4	Nuclear Boiler System Instrumentation Subsystem.....	5-3
5.1.5	Main Steam Flow Restrictors .....	5-4
5.1.6	Reactor Isolation Valves and Main Steam Containment Isolation Valves.....	5-5
5.1.7	Control Rod Drive System .....	5-5
5.1.8	Reactor Water Cleanup System.....	5-5
5.1.9	Isolation Condenser System .....	5-5
5.1.10	Piping and Instrumentation Schematics .....	5-6
5.1.11	Elevation Schematics .....	5-6
5.1.12	References .....	5-6
5.2	Material Requirements Associated with the NBS and RCPB .....	5-11
5.2.1	Basis of Material Selection and Component Fabrication .....	5-11
5.2.2	Material Properties.....	5-13
5.2.3	Overview of Reactor Pressure Vessel and Pressure Boundary Components .....	5-14
5.2.4	Degradation Mechanisms for Reactor Pressure Vessel and Pressure Boundary Components .....	5-15
5.2.5	Overview of Core Structural Component Materials .....	5-16
5.2.6	Overview of Design and Fabrication Process for SCC Controls in Reactor Core Components .....	5-18
5.2.7	Gasket, Seal, and Fastener Materials .....	5-20
5.2.8	References .....	5-20
5.3	Nuclear Boiler System .....	5-26
5.3.1	Nuclear Boiler System Configuration .....	5-26
5.3.2	NBS Classification of SSC .....	5-29
5.3.3	NBS Functions.....	5-29
5.3.4	System Operations .....	5-30
5.3.5	Refueling Mode Operation .....	5-32
5.3.6	Testing and Maintenance.....	5-32

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

5.3.1	References .....	5-33
5.4	Reactor Pressure Vessel .....	5-34
5.4.1	Major Reactor Internal Components .....	5-34
5.4.2	Core Components (Control Rods and Nuclear Instrumentation) .....	5-34
5.4.3	Core Support Structures .....	5-35
5.4.4	Chimney .....	5-37
5.4.5	Chimney Head and Steam Separator Assembly .....	5-37
5.4.6	Steam Dryer Assembly .....	5-37
5.4.7	Feedwater Spargers .....	5-37
5.4.8	Internal Water Piping .....	5-37
5.4.9	Reactor Pressure Vessel Head Vent .....	5-38
5.4.10	Nuclear Instrumentation Tubes and Stabilizers.....	5-38
5.4.11	Surveillance Sample Holders.....	5-38
5.4.12	References .....	5-38
5.5	Reactor Coolant Pumps or Recirculation Pumps .....	5-40
5.6	Reactor Coolant Piping.....	5-41
5.6.1	References .....	5-41
5.7	Reactor Pressure Control .....	5-42
5.7.1	System and Equipment Functions .....	5-42
5.7.2	Safety Design Bases .....	5-42
5.7.3	Description .....	5-42
5.7.4	Interfaces with Other Equipment or Systems .....	5-42
5.7.5	System and Equipment Operation .....	5-42
5.7.6	Instrumentation and Control.....	5-42
5.7.7	Monitoring, Inspection, Testing, and Maintenance .....	5-43
5.7.8	Radiological Aspects .....	5-43
5.7.9	Performance and Safety Evaluation.....	5-43
5.7.10	References .....	5-44
5.8	Isolation Condenser System.....	5-45
5.9	Reactor Pressure Vessel Component Supports and Restraints .....	5-46
5.9.1	Reactor Pressure Vessel Pedestal .....	5-46
5.9.2	Reactor Pressure Vessel Pedestal Safety Basis.....	5-46
5.9.3	Reactor Pressure Vessel External Supports .....	5-46
5.9.4	Reactor Pressure Vessel Internal Supports .....	5-47

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

5.9.5	Materials .....	5-48
5.9.6	Interfaces with Other Equipment or Systems .....	5-48
5.9.7	Monitoring, Inspection, Testing, and Maintenance .....	5-48
5.9.8	Radiological Aspects .....	5-48
5.9.9	Performance and Safety Evaluation.....	5-48
5.9.10	References .....	5-49
5.10	Nuclear Boiler System Connected Valves .....	5-50
5.10.1	Reactor Isolation Valves .....	5-50
5.10.2	Main Steam Containment Isolation Valves.....	5-50
5.10.3	Safety Design Bases .....	5-51
5.10.4	Materials.....	5-51
5.10.5	Interfaces with Other Equipment or Systems .....	5-51
5.10.6	Instrumentation and Control.....	5-51
5.10.7	Monitoring, Inspection, Testing, and Maintenance .....	5-52
5.10.8	Radiological Aspects .....	5-52
5.10.9	Performance and Safety Evaluation.....	5-52
5.10.10	References .....	5-52
5.11	Access and Equipment Requirements for Inspection and Maintenance.....	5-55
5.11.1	Accessibility .....	5-55
5.11.2	Reactor Coolant Pressure Boundary Components.....	5-55
5.11.3	Reactor Coolant System and Connected Systems.....	5-55
5.11.4	Examination Categories and Methods .....	5-56
5.11.5	Inspection Intervals.....	5-58
5.11.6	Provisions for Evaluating Examination Results .....	5-59
5.11.7	System Pressure Tests.....	5-59
5.11.8	Surveillance Program Requirements and Overview .....	5-60
5.11.9	Program and Milestone Implementation.....	5-61
5.11.10	References .....	5-61
5.12	Reactor Auxiliary Systems.....	5-62
5.12.1	Chemical and Inventory Control Systems for the Reactor Coolant.....	5-62
5.12.2	Reactor Water Cleanup System.....	5-62
5.12.3	High Point Vents of the Nuclear Boiler System .....	5-63
5.12.4	Control Rod Drive System .....	5-63



**LIST OF TABLES**

Table 5.2-1: Typical Materials for Reactor Pressure Vessel Components .....	5-21
Table 5.2-2: Typical Compositions of Low Alloy Steel and Carbon Steel Materials .....	5-22
Table 5.2-3: Typical Reactor Internal Components and Materials .....	5-23
Table 5.2-4: Typical Compositions: Standard BWR-300 Materials and Special Compositional Controls for Toughness and IASCC Resistance .....	5-24

## LIST OF FIGURES

Figure 5.1-1: Nuclear Boiler System Input – Output Diagram .....	5-7
Figure 5.1-2: Nuclear Boiler System Simplified Diagram (Located in the Reactor Building) .....	5-8
Figure 5.1-3: Nuclear Boiler System Simplified Diagram (Located in Turbine Building) .....	5-9
Figure 5.1-4: Reactor Coolant Pressure Boundary .....	5-10
Figure 5.2-1: The Three Necessary Factors to Produce Stress Corrosion Cracking .....	5-25
Figure 5.4-1: BWRX-300 Reactor Pressure Vessel and Internals .....	5-39
Figure 5.10-1: Reactor Pressure Vessel Isolation Valve Assembly (Example) .....	5-54

## 5.0 NUCLEAR BOILER SYSTEM AND ASSOCIATED SYSTEMS

This chapter describes the BWRX-300 Nuclear Boiler System (NBS) which generates and delivers steam to the turbine for power generation. The NBS and interfacing systems form the Reactor Coolant Pressure Boundary (RCPB). The Reactor Pressure Vessel (RPV) forms the major portion of the RCPB and contains the path for reactor coolant flow through the fuel and generates steam. Further, this chapter addresses the functional and structural integrity aspects of the various NBS Structures, Systems, and Components (SSC) that are designed with robustness, quality, independence, redundancy, and diversity to maintain adequate reactor coolant inventory during Normal Operation, Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs). The NBS and RCPB design utilize the Safety Strategy Defence-in-Depth (D-in-D) Defense Lines (DLs) as described in Chapter 3, Section 3.1 to meet Canadian regulatory requirements and expectations.

The portion of the NBS and system interfaces located in the Reactor Building (RB) is illustrated in Figure 5.1-2 and described in Chapter 5. The portion of NBS located in the Turbine Building is presented in Figure 5.1-3 and discussed in detail in Chapter 10, Section 10.4.

The functional aspects of the fuel and reactivity control within the RPV are discussed in Chapter 4. The BWRX-300 design includes Engineered Safety Features (ESFs) that mitigate the consequences of AOOs or postulated DBAs without any core damage. The ESFs associated with the NBS interfacing systems are described in Chapter 6, Section 6.2.

### 5.1 Summary Description

The BWRX-300 Nuclear Boiler System (NBS) is comprised of three primary subsystems: Reactor Pressure Vessel (RPV), Main Steam (MS), and NBS instrumentation. In addition to these subsystems, the NBS also provides RPV head venting and RPV head flange seal leak detection features. Within the NBS, the SC1 Reactor Isolation Valves (RIVs) are mechanically redundant and Instrumentation and Control (I&C) diverse. These RIVs interface with the following subsystems and systems: Main Steam (MS), RPV Head Vent, Isolation Condenser System (ICS) Supply, ICS Return, Condensate and Feedwater Heating System (CFS), and Reactor Water Cleanup System (CUW). The Nuclear Boiler System and associated systems are designed in accordance with requirements of ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components" (Reference 5.1-1) and ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1- Subsection NCD-Class 2 and Class 3 Components" (Reference 5.1-2).

The BWRX-300 NBS and interfacing systems are a unique design which mitigates LOCAs that could impact Reactor Vessel Water Levels by providing double RIVs directly flanged to the integral RPV nozzles. The RIVs are closer to the RPV compared to predecessor Boiling Water Reactor (BWR) designs (i.e., there is no piping between the RPV and RIVs). Figure 5.1-1 provides the input-output block diagram that lists all intersecting systems with the NBS and describes the interfaces between systems. The portion of the NBS (Main Steam Section) from the Seismic Interface Resistant (SIR) to the Turbine Stop Valves (TSV), Turbine Bypass Valves (TBV), and other steam loads is addressed in Chapter 10, Section 10.4.

The NBS is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, and reduce the number and size of RPV nozzles as compared to predecessor designs. All nozzles are located significantly above the top of active fuel (TAF). The relatively large RPV volume, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This ensures the reactor water level is maintained above TAF and fuel cladding temperature is maintained within normal operating temperature range following

transients involving Feedwater (FW) flow interruptions or LOCAs. These design safety features preserve reactor coolant inventory to ensure that adequate core cooling is maintained.

Innovative features related to BWRX-300 are the partitionless RPV chimney region flow characteristics (see Subsection 5.4.3); RIV qualification (see Section 5.10) and the RPV water level measurement in the core region (see Subsection 5.4.9). The large RPV volume enhances safety by reducing the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its normal heat sink. The containment isolation function is addressed separately in Chapter 6, Subsection 6.3.4.

Figure 5.1-2 provides the portion of the NBS that is described in Chapter 5 and shows the double isolation RIVs of those systems that intersect the reactor pressure vessel. Additionally, Figure 5.1-4 provides the extent of the RCPB.

The relevant CNSC Regulatory Documents (REGDOCs), codes, standards have been identified and description of their applicability to design and construction activities as part of establishing the LTC Licensing Basis is documented in NK054-REP-01210-00137, "DNNP License to Construct Regulatory Documents, Codes & Standards" (Reference 5.1-3).

#### **5.1.1 Reactor Coolant Pressure Boundary**

The RCPB (see Section 5.3), contains all pressure-retaining components such as:

- The Reactor Pressure Vessel (RPV)
- The integral Reactor Isolation Valve (RIV)
- The piping from the outboard RIVs up to each respective containment isolation valve (CIV)
- The outboard CIVs for Main Steam (MS), Isolation Condenser System (ICS) to Shutdown Cooling System (SDC), ICS to Boron Injection System (BIS), Reactor Water Cleanup System (CUW), and Condensate and Feedwater Heating System (CFS)
- The Isolation Condenser System (ICS) steam supply lines, isolation condensers, purge lines, and condensate return lines
- The head vent piping to the Main Steam Line (MSL) and boundary valves for the Quench Tank
- The reactor instrumentation lines
- The control rod drive interface flanges/housings
- The nuclear instrumentation interface flanges/housings

The RCPB is presented in Figure 5.1-4.

#### **5.1.2 Reactor Pressure Vessel and Appurtenances**

The RPV is used as a pressure-retaining barrier of primary coolant in a BWRX-300 reactor. The vessel contains the light water coolant and moderator and forms the path for core recirculation flow. The vessel also contains nuclear fuel, saturated steam, fuel supporting structures, and internals required for safe operation of the reactor. Production of thermal energy from the nuclear fission process takes place within the vessel. The above function requires that the vessel be a Safety Class (SC) 1 vessel, designed to meet the requirements of ASME BPVC Section III, Division 1, Subsection NB (see Figure 5.4-1 for the major components contained in the RPV). A more detailed description of the reactor pressure vessel and its major components is provided in Section 5.4.

The RPV has integral isolation valves attached directly to and are considered appurtenances of the RPV. The SC1 RIVs are mechanically redundant and Instrumentation and Control (I&C) diverse.

The RPV design is such that the physical level of all nozzles are located significantly above TAF. In addition to multiple system interfaces via the RPV nozzles, a flanged connection is provided in the RPV bottom head for each Fine Motion Control Rod Drive (FMCRD) and each neutron monitoring detector.

The RPV head vent subsystem (SC1) includes piping internal to the RPV head, two RIVs in-series flange-mounted on the reactor, and piping to one of the MS lines and the Quench Tank, both connections located inside the Primary Containment System (PCS).

The major safety consideration for the reactor pressure vessel is the ability of the vessel to function as a radioactive material barrier under Normal, AOO, DBA, and DEC plant states as described in Chapter 3, Section 3.1.

### **5.1.3 Main Steam Subsystem**

The Main Steam (MS) subsystem consists of two steam lines from the discharge flange of the outboard main steam reactor isolation valves (MSRIVs) to the Turbine Stop Valves (TSVs), the Turbine Bypass Valves (TBVs), the Main Steam Line (MSL) drains, and other load isolation/maintenance valves. The supply lines to these loads, all connecting branch lines up to and including their respective isolation valves, and all associated piping supports are also part of the MS subsystem.

The MSL drains remove any condensate from the MSLs to the main condenser during startup, low power operation, normal power operation, and shutdown. A reduction in power to a low-level, results in the automatic opening of the air-operated drain line valves, thereby establishing drain flow to the main condenser.

The portion of the MS piping from the outboard MSRIVs to the Seismic Interface Restraint (SIR) outboard of the main steam containment isolation valves (MSCIVs) is SC1 and designed to meet the requirements of ASME BPVC Section III, Division 1, Class 2. The remaining MS piping from the SIR to the TSVs, the TBVs, the MSL drains, and other load isolation/maintenance valves is designed to meet the requirements of ASME B31.1, "Power Piping" (Reference 5.1-4) as described in Chapter 10, Section 10.4.

### **5.1.4 Nuclear Boiler System Instrumentation Subsystem**

The NBS is the source of the reactor water level, reactor pressure, temperature, and steam flow information. The NBS instrumentation consists of sensors to measure and monitor parameters including reactor coolant pressure, flow, temperature, water level, and reactor power. Additionally, there are sensors to measure and monitor MSL pressure, flow, drain temperature, RPV metal temperatures, RPV head flange seal pressure (leak detection), RPV head venting, steam tunnel temperature, and Turbine Building (TB) temperatures near the MSLs for leak detection purposes.

Neutron flux detectors for reactor power provide signals to the C10 Power Range Neutron Monitoring system (PRNM) and the C20 Wide Range Neutron Monitoring system (WRNM) (refer to Chapter 7, Subsection 7.3.3.2 for detailed discussion). The non-neutron flux sensors provide data to instruments or signal conditioning and data acquisition equipment that are typically mounted on instrument racks outside of containment. The instruments are appropriately separated and divided into Defense Lines (DLs) and safety classifications.

The NBS also has control requirements for valves (RIVs, CIVs, and drain valves) that are repositioned as the plant is brought from cold to full power or that respond to unplanned transients. Control design requirements include that SC1 equipment is optically isolated from the SC3 C20 Plant Automation System (PAS). Therefore, provisions are made to operate the valves manually by the reactor operator as prompted by the PAS or C10 Visual Display Units (VDUs). The SC3 valves may be operated by the PAS.

The nuclear boiler controller logic architecture (see Chapter 7, Subsection 7.2.1) uses Triple Modular Redundant technology and is dual ported to the plant nuclear segment network. As with all Triple Modular Redundant controllers, extensive hardware and software diagnostics are provided to provide for operator monitoring and alarms. There is no connection between DL3/SC1 and the nuclear segment controllers and no sensors or actuators are in common.

#### **5.1.4.1 Reactor Pressure Vessel Level Instrumentation**

Normal operations (DL1) for RPV water level is controlled by reactor level control system (as described in Chapter 7, Section 7.3). Water level measurement by differential pressure is also used for transient/AOO (DL2) and DBA (DL3) monitoring. The Differential Pressure (dP) instrumentation is setup to provide a more accurate narrow range (NR) measurement as well as a less accurate, but otherwise sufficient wide range (WR) measurement. The BWRX-300 is designed to have no RPV penetrations below top of active fuel + 4.0 m (TAF+4m). This is a plant design requirement to preserve sufficient condensate inventory for analyzed events and accidents. Because there are not any operator actions associated with water level between TAF and TAF+4m, there are no water level measurements available in this range.

In the case of DL4 (i.e., Beyond Design Basis Accident (BDBA) and DEC) events, Gamma Thermometers (GTs) will be utilized for the fuel zone water level measurements. The GT is an extension of the local power range monitoring (LPRM) calibration system within the core. The water level inside of the core will show adequate coverage for core cooling without the need for an extra system in the downcomer or on the RPV Outside Diameter (OD). It is generally considered that the level inside of the core provides the best information.

The water inventory in the RPV is credited for normal operations and AOOs as described in Chapter 15, Subsection 15.5.3 (decrease/increase in reactor coolant inventory). The reactor level control remains unaffected by these failures and is able to maintain level. As demonstrated in Chapter 15, Section 15.3, the RCPB maintains sufficient coolant inventory for core cooling as long as reactor level is maintained at or above top of the active fuel (the derived acceptance criteria for AOOs).

#### **5.1.4.2 Nuclear Boiler System Pressure, Flow, and Temperature**

The other NBS instrumentation functions are to measure reactor coolant pressure, flow, and temperature. More details are provided in Subsection 5.3.1.3.

#### **5.1.5 Main Steam Flow Restrictors**

The main steam flow restrictors are located in the RPV MS nozzles. They perform functions in various safety classifications and corresponding defence levels. The SC1 functions are to limit the overall steam flow through one steam line when the steam flow exceeds preselected operational limits (if a large break of a MSL is sensed) and the corresponding MSRIVs and MSCIVs have not closed. The other SC1 function is to provide that same sensing function to the SC1 MSRIV and MSCIV isolation system. The SC2 steam flow function also supplies the same sensing functions for MSRIV and MSCIV isolation in addition to the already discussed SC1 isolation function. The SC3 steam flow function supports the overall reactor water level controller that maintains target water level as one of the inputs to that controller.

The main steam flow restrictor provides low-pressure sensing taps for instrumentation lines for differential pressure transmitters that are used to measure steam flow. Main steam flow is measured through the MS nozzle venturis using Bernoulli's equation of differential pressure to derive the MS mass flow.

The main steam flow restrictor is designed in accordance with the Report of ASME Research Committee on Fluid Meters and has no moving parts. Required plant duty cycles associated with MS flow restrictors and other ASME BPVC Section III, Division 1, Class 1 components are provided in Chapter 3, Section 3.6.

#### **5.1.6 Reactor Isolation Valves and Main Steam Containment Isolation Valves**

A novel feature of the BWRX-300 is the integral RIVs that are directly flanged to nozzles on the reactor vessel instead of welding a pipe to the vessel nozzle and then connecting the isolation valve to the pipe at some distance from the reactor. As appurtenances to the RPV, the RIVs allow large pipe break LOCAs to be quickly isolated, thereby, retaining the fluid inventory in the vessel for core cooling. Each RPV nozzle with an internal bore greater than DN20 is required have two integral RIVs flanged directly to the nozzle. The main function of the isolation valves is to close and mitigate the effects of LOCAs from large and medium pipe breaks. The RPV automatic isolation concept consists of two RIVs in series and is single failure proof. With each of the RIVs independently able to isolate the line. RIVs are placed inboard of each large and medium sized pipe that is connected to the RPV.

The BWRX-300 design includes one Main Steam Containment Isolation Valve (MSCIV) for each main steam line outside the containment. If a main steam line break occurs inside the containment, closure of all outside Containment Isolation Valves (CIVs) seals the containment.

Additionally, all RIVs with the exception of the ICSRIVs and ICRRIVs will automatically close to isolate the RCPB at the RPV when a pipe break occurs either inside or outside of containment. This action limits the loss of reactor coolant and the release of radioactive materials. Further discussion on operation of the RIVs and MSCIVs is provided in Section 5.10.

#### **5.1.7 Control Rod Drive System**

The control rod drive (CRD) contains components which form part of the reactor coolant pressure boundary, along with components which are important to safety to shut down the reactor. Those portions of the CRD are classified as DL3/SC1.

Operation of the CRD system is further discussed in Chapter 4, Section 4.6.

#### **5.1.8 Reactor Water Cleanup System**

The Reactor Water Cleanup System (CUW) provides blowdown-type cleanup flow for the RPV during the reactor power operating mode. CUW also provides an overboarding flowpath (i.e., excess RPV coolant inventory discharge for level control) to the condenser hotwell or Liquid Waste Management System (LWM) directly from the RPV lower region.

Detailed design description of CUW is provided in Chapter 9A, Subsection 9A.2.2.

#### **5.1.9 Isolation Condenser System**

The Isolation Condenser System (ICS) removes decay heat after any reactor isolation and shutdown event during power operations when the main condenser is not available. The ICS is connected to the RPV by steam supply piping and condensate return piping. The large ICS heat removal capacity limits increases in RPV pressure and maintains the RPV pressure at an acceptable level. The Isolation Condensers (IC) condense steam from the RPV and transfer thermal energy by convection to the ICS pool water.

ICS is responsible for protecting the integrity of the RCPB in the event of a large line break LOCA that results in the isolation of the RPV.

During shutdown modes, ICS will perform the RPV isolation function for a line break within the Shutdown Cooling (SDC) system by isolating the ICS-SDC interface valves. This function prevents an ICS train from being rendered inoperable by what otherwise would be an RIV closure in the case of an SDC LOCA. As described in Chapter 15, Subsection 15.5.4.6, ICS removes decay heat generated in the core and does not require coolant injection to mitigate pipe breaks and transients.

The ICS is an Emergency Core Cooling System (ECCS) that also provides RPV overpressure protection for which further details can be found in Chapter 6, Section 6.2.

#### **5.1.10 Piping and Instrumentation Schematics**

Piping and instrumentation schematic diagrams covering the systems included within NBS and connected systems are presented as follows:

- Nuclear Boiler System (Figure 5.1-2 and Figure 5.1-3)
- Isolation Condenser System (Chapter 6, Figure 6.2-6)
- Reactor Water Cleanup System (Chapter 9, Figure 9A.2.2-1)
- Shutdown Cooling System (Chapter 9, Figure 9A.2.3-1)

#### **5.1.11 Elevation Schematics**

The elevation schematic showing the principal features of the reactor and connecting systems in relation to the containment are provided in Chapter 1, Figure 1.7-1.

#### **5.1.12 References**

- 5.1-1 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 5.1-2 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 5.1-3 NK054-REP-01210-00137, "DNNP License to Construct Regulatory Documents, Codes & Standards," Ontario Power Generation.
- 5.1-4 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.



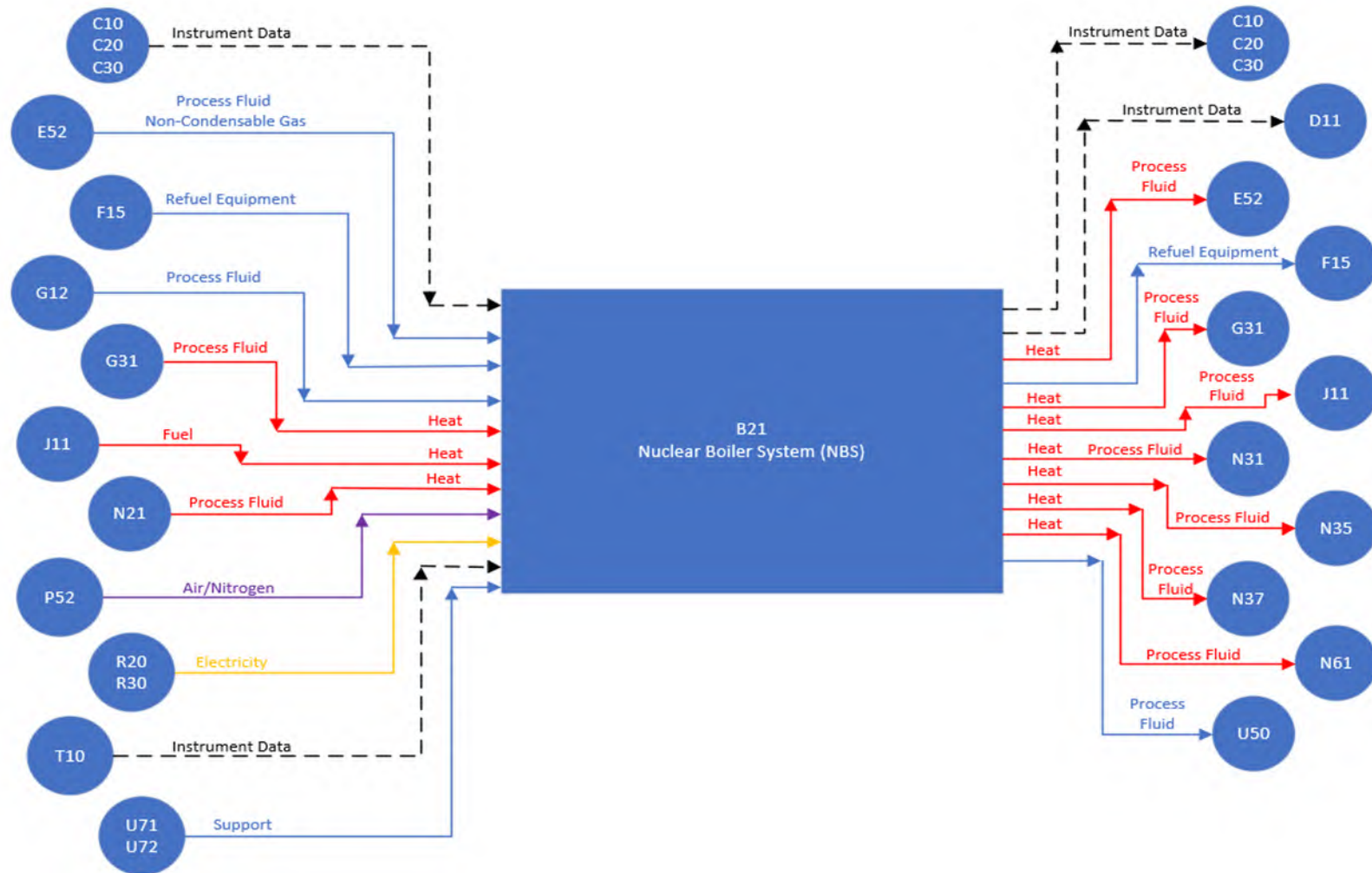
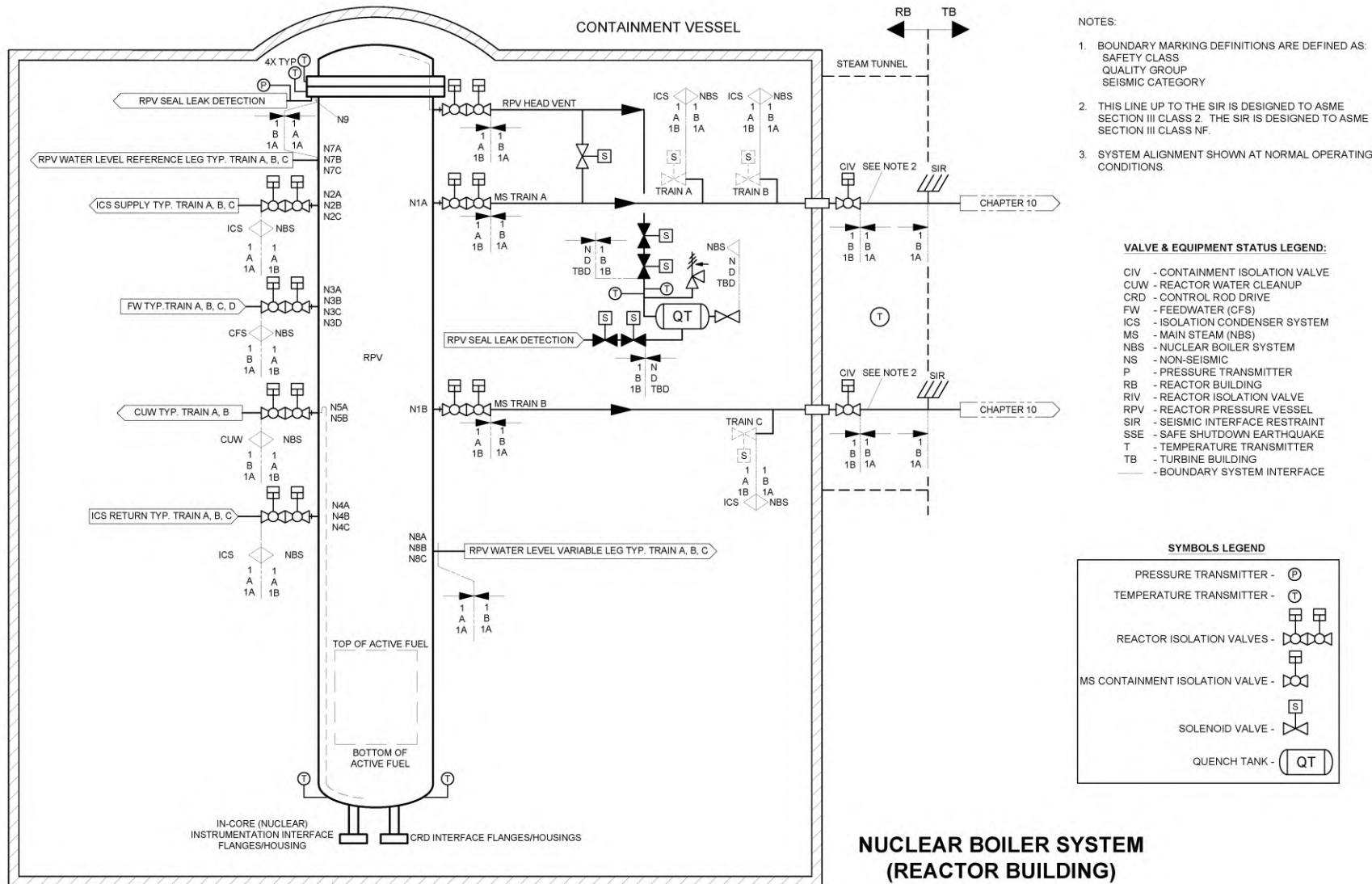
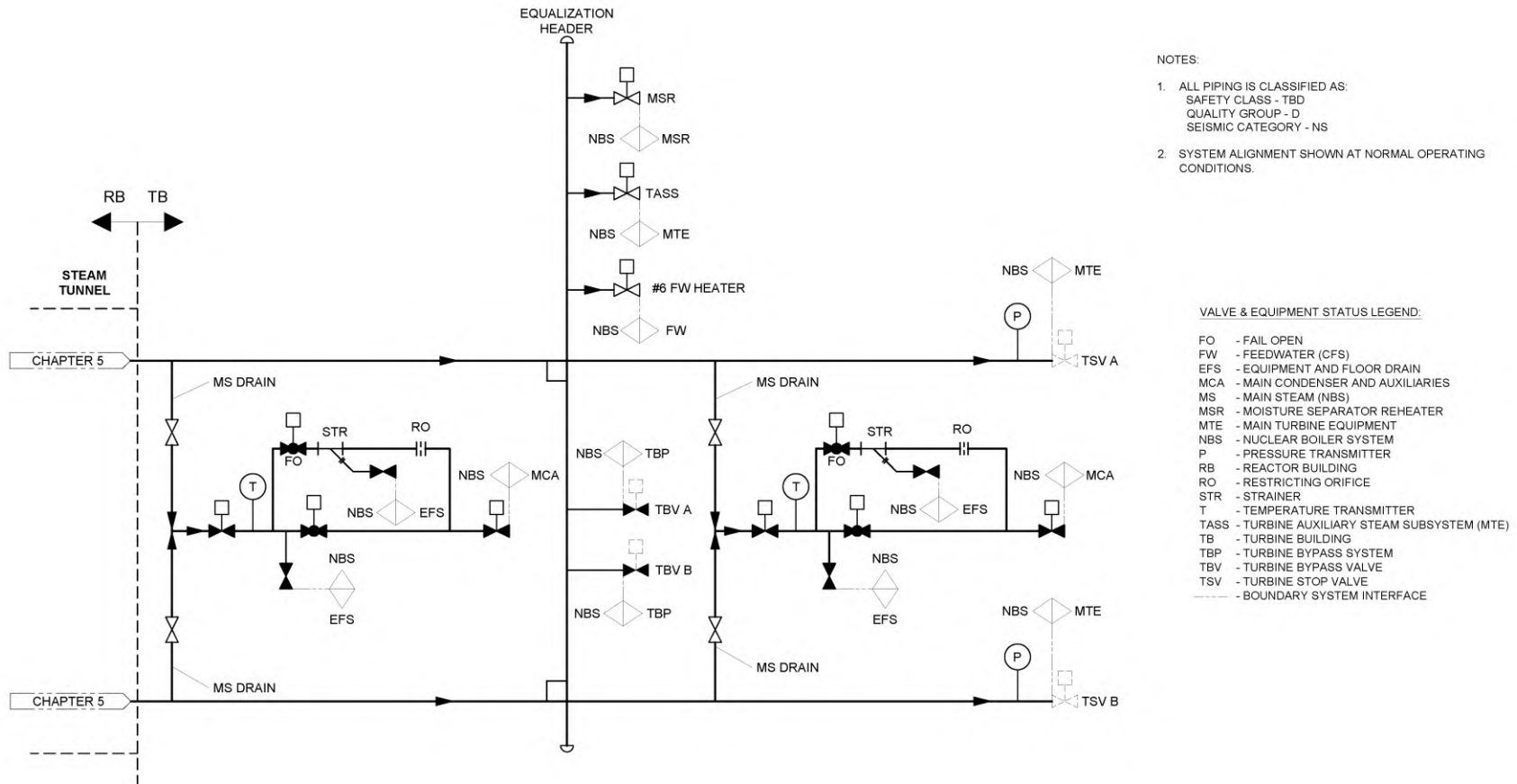


Figure 5.1-1: Nuclear Boiler System Input – Output Diagram

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION



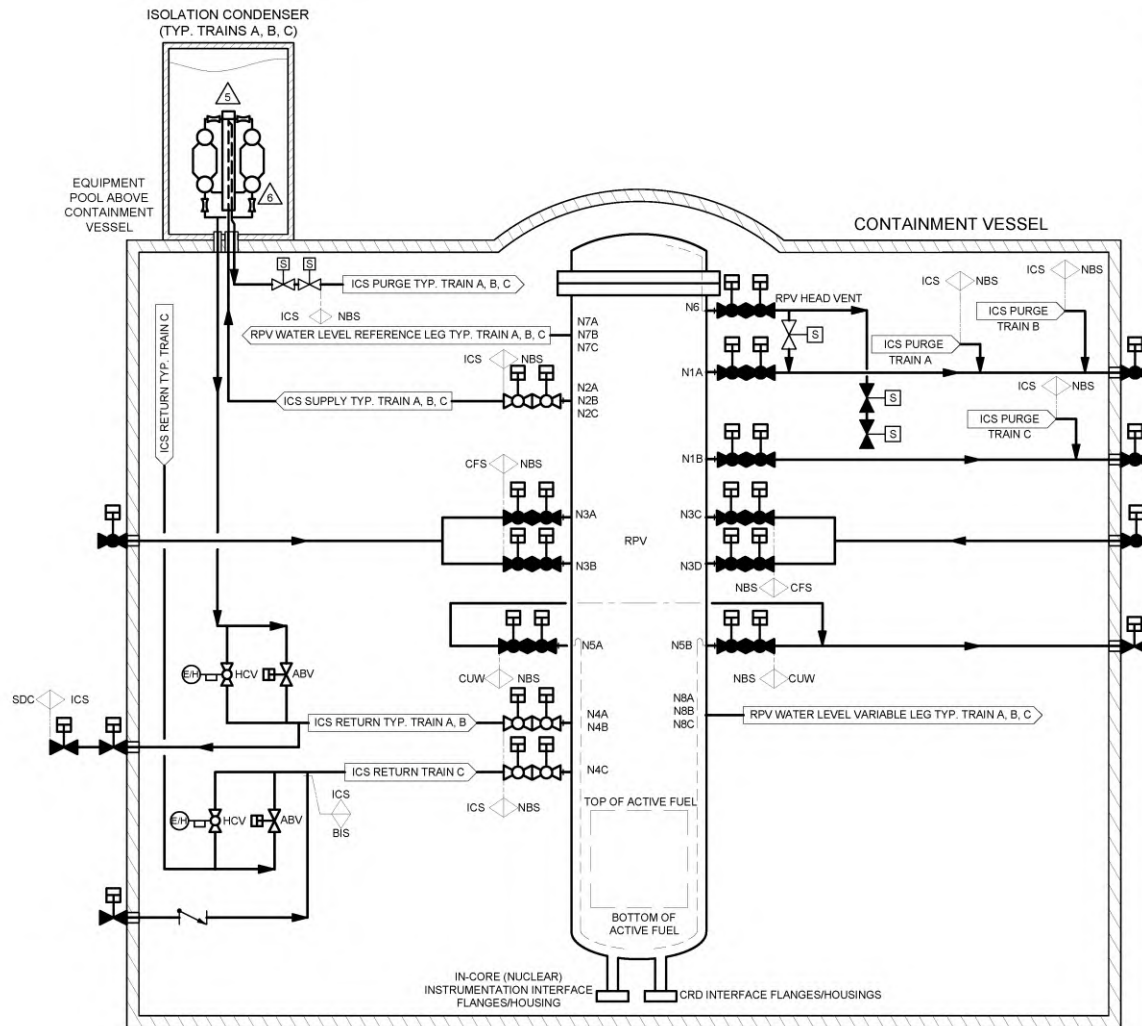
NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION



**NUCLEAR BOILER SYSTEM  
(TURBINE BUILDING)**

**Figure 5.1-3: Nuclear Boiler System Simplified Diagram (Located in Turbine Building)**

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION



NOTES:

1. EQUIPMENT, PIPING, & VALVES WITH BOLDED LINES DEPICT THE REACTOR COOLANT PRESSURE BOUNDARY.
2. ENTIRE REACTOR COOLANT PRESSURE BOUNDARY IS CLASSIFIED AS SAFETY CLASS 1 (SC1).
3. THE RPV LEVEL INSTRUMENTATION SENSING LINES ARE CLASSIFIED AS ASME BPVC SECTION III CLASS 2 PER ASME BPVC SECTION III PARAGRAPH NB-3630(D).
4. THE PIPING OUTBOARD OF THE RIV'S ARE CLASSIFIED AS ASME BPVC SECTION III AS CLASS 2, EXCLUDING ICS PIPING UP TO ISOLATION CONDENSERS.

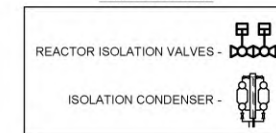
5. THE IC IS DESIGNED TO BOTH ASME SECTION III, SUBSECTION NB, CLASS 1 AND ASME SECTION III, SUBSECTION NCD, ASME CLASS 2 AND 3. THE STEAM SUPPLY AND PURGE LINE AND THEIR ASSOCIATED COMPONENTS ARE DESIGNED TO CLASS 1 THROUGH THE CONTAINMENT BOUNDARY UP TO AND INCLUDING THE FOUR FLOW RESTRICTORS.

6. THE IC IS DESIGNED TO ASME CLASS 2 DOWNSTREAM OF THE FLOW RESTRICTORS, INCLUDING THE UPPER HEADER, CONDENSER TUBES, LOWER HEADER, VENT LINES, AND THE TWO CONDENSATE DRAIN LINES UP TO THE POINT WHERE THEY RECOMBINE INTO A SINGLE LINE. THE IC DESIGN RETURNS TO ASME CLASS 1 STARTING AT THE CONDENSATE TEE BACK THROUGH THE CONTAINMENT PENETRATION THROUGH THE ICRRVS INTO THE REACTOR.

VALVE & EQUIPMENT STATUS LEGEND:

- ABV - BLOCK VALVE (PNEUMATIC ACTUATED)
- BIS - BORON INJECTION SYSTEM
- CRD - CONTROL ROD DRIVE
- E/H - ELECTRO-HYDRAULIC
- HCV - BLOCK VALVE (HYDRAULIC ACTUATED)
- ICS - ISOLATION CONDENSER SYSTEM (ICS)
- RIV - REACTOR ISOLATION VALVE
- RPV - REACTOR PRESSURE VESSEL
- S - BLOCK VALVE (SOLENOID ACTUATED)
- - BOUNDARY SYSTEM INTERFACE
- - INTERNAL RPV PIPING
- - CUW PIPING

SYMBOLS LEGEND



REACTOR COOLANT  
PRESSURE BOUNDARY

Figure 5.1-4: Reactor Coolant Pressure Boundary

## **5.2 Material Requirements Associated with the NBS and RCPB**

The material and process control requirements for the BWRX-300 components have been defined to ensure the reliability of the plant operations through its design life, by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials, specifically from Intergranular Stress Corrosion Cracking (IGSCC) through material chemistry, heat treatment, contamination, and material processes controls.

The BWRX-300 NBS employs proven BWR materials and processes which have been refined to meet reactor requirements.

The basic design principle for materials selection is for the materials to be capable of maintaining reliable operation of plant systems and equipment during the design life of each component. In principle, carbon steels including atmospheric corrosion resistant carbon steels and low alloy steels are used as basic materials. In order to minimize levels of radiation from corrosion products, reactor internals are made of austenitic stainless steel. Materials selection and controls address materials degradation issues in the reactor system such as Stress Corrosion Cracking (SCC), general corrosion, and Flow Accelerated Corrosion (FAC).

SCC is considered the dominant form of corrosion in a BWR. Significant efforts through the years have been expended to understand it and control SCC. It is a complex phenomenon that involves mechanical, electrochemical, and metallurgical factors. SCC is usually characterized by localization of the cracked region near welds or near regions of high surfaces strains or stresses. For SCC to occur, it requires three necessary components to be simultaneously present. The elimination of any one of these factors, or reduction of one of these factors below a threshold level eliminates the risk of SCC. These three necessary conditions for SCC as schematically shown in Figure 5.2-1 are (1) Susceptible material; (2) Tensile Stress (applied or residual) and (3) Corrosive environment.

The different degradation mechanisms that potentially affect the integrity of the construction materials that are used in the BWRX-300 are discussed in the subsequent sections.

### **5.2.1 Basis of Material Selection and Component Fabrication**

The structural materials used for the RPV and its attachments, the core support structure, internals, and the piping in the BWRX-300 design are selected and fabricated to meet the design requirements of the reactor components while maintaining high resistance to known degradation mechanisms during plant operation. Seals, gaskets, and fasteners are selected based on experience and industry acceptance. To monitor material degradation, guidelines for surveillance programs for BWRX-300 are informed by ASME and industry guidelines.

The BWR uses high purity water as the fluid to transfer thermal energy. The quality of water to which the component is exposed, to a large extent, determines the reliability and lifetime of that component. Generally, corrosion effects stem from the combination of materials selected, the stresses or duty applied, and the environment to which the component is exposed. The extent of materials corrosion from its environment is a function of impurity and its concentration, temperature, and time. Water of required quality is achieved by the use of appropriate water treatment equipment and systems, as well as the application of appropriate materials in system construction.

The material selections for the BWRX-300 reactor system are made without taking credit for the IGSCC suppression benefits of Hydrogen Water Chemistry. However, it is recognized that Hydrogen Water Chemistry does provide an additional margin of IGSCC resistance for most components exposed to reactor water. Additionally, Hydrogen Water Chemistry in conjunction



NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

with noble metal technology, e.g., NobleChem™ or noble metal alloys, provides the ability to achieve Hydrogen Water Chemistry benefits while limiting radiation dose in the steam lines.

The BWRX-300 builds on a long history of material experience and refinement of material alloy chemistry to assure the best characteristics to preclude long-term material degradation. These materials include wrought alloys as well as weld metals used in the construction. Given the decades of experience as well as the strength of utility interactions through EPRI and research efforts by GEH and its licensees, there is high confidence that degradation will be very limited if not eliminated for the life of the plant. In addition, water chemistry environment controls to address potential degradation mechanisms are being implemented.

Subsection 5.2.3 covers low alloy steels and carbon steels with the associated weld metals used for the RPV, piping and vessel attachments (including nozzles and bolting). Subsection 5.2.4 includes a discussion of the potential degradation mechanisms. Subsection 5.2.5 discusses austenitic stainless steel and nickel-based alloys used for reactor internal components such as the core shroud, chimney, core support structure and control rod and instrumentation hardware as well as several piping systems. Relevant degradation mechanisms are discussed in Section 5.2.6, in particular SCC are discussed for these austenitic components.

All pressure boundary material specifications used in the BWRX-300 are as per ASTM/ASME standards. Pressure boundary materials are selected with consideration to environment compatibility with an emphasis for limiting the effects of corrosion in the reactor coolant environment due to contaminants and radiolytic products. This is especially true for ferritic low alloy steels and carbon steels.

In the case of stainless steels, the major environment-related materials performance issue encountered to date in the RCPB of BWRs has been IGSCC of sensitized austenitic stainless steel. IGSCC has been discovered in sensitized material adjacent to welds in Type 304L and Type 316L stainless steel piping systems. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon. Remedial measures have been developed and implemented for the BWRX-300. IGSCC resistance is achieved through the use of IGSCC resistant materials such as Type 316L stainless steel with controlled composition. Welding controls are also implemented to ensure that welding filler materials, welding techniques and controlled post-weld heat treatment are utilized to minimize the potential for IGSCC. Additionally, weld locations in the reactor internals welds are strategically placed to limit further sensitization from irradiation. Water chemistry is also tightly controlled to minimize the existence of an environment that could promote IGSCC.

The BWRX-300 water chemistry program follows the best industry water chemistry guidelines and it is designed to analyze and monitor system chemistry for trending with alarm notification so actions can be taken to stay within operating specifications. This supports minimizing corrosion from chemical contaminants and monitoring chemical additives used to limit corrosion and radiation buildup.

Different materials are used to manufacture the reactor components, the reactor pressure vessel, piping, and reactor internals. Additionally, the typical materials used for gaskets, seals, and fasteners will vary dependent on the location and environmental conditions. For the primary components, the key materials, typical composition limits and ASME/ASTM specifications along with the fabrication processes are used to meet the requirements for the BWRX-300 application to assure resistance to different types of degradation. GEH specified materials and process controls are used to define the requirements related to service conditions. The ASME Code specifications and material alloy chemistry, define their chemical, physical and mechanical properties, that produce resistance to corrosion, dimensional stability, and desired toughness in

the fabricated components. Key microstructural features resulting from the material fabrication processing methods are also critical to achieving resistance to degradation over the plant lifetime.

### **5.2.2 Material Properties**

Nuclear power plant components may be exposed to environmental factors whose single and combined cumulative effects cannot be predetermined for the operating lifetime of the plant. The most important environmental factors are stress, high temperature, irradiation, operating transients, hydrogen absorption, corrosive attack, vibration, and fretting, with all these factors depending on time and operating history. These environmental factors can result in changes to material properties caused by irradiation or thermal embrittlement, corrosion fatigue and potentially lead to the initiation and growth of flaws. Therefore, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness to minimize the possibility of brittle fracture of the NBS and RCPB components.

Materials used in the construction of the portion of the NBS that provide a safety class function or materials that are in contact with the process fluid, are selected based on suitability for the service based on the requirements of the applicable codes and standards. NBS piping is designed using materials selected to prevent erosion caused by high local steam flow velocity.

Materials used for the PCS, penetration piping and the associated supports are designed in accordance with the rules and requirements of ASME BPVC, "Section II, Materials" (Reference 5.2-1).

Where applicable, materials are impact tested in accordance with ASME BPVC Section III, Division 1, Class 1, and Class 2 components, using instruments and equipment certified to meet the same ASME BPVC requirements.

The RPV also complies with the toughness requirements for Class 1 vessels per ASME BPVC Section III. This includes plates, forgings, weld material, and the heat affected zones. Minimum toughness values at various locations of the RPV are confirmed through Charpy V-notch testing, as per ASME BPVC Section III. Charpy curves and fracture toughness specimens are used for evaluation of radiation effects on embrittlement with operating time. Consideration has been given to the effects of irradiation on beltline fracture toughness by controlling the chemical composition of vessel beltline materials.

Safety class pressure boundary components are designed and manufactured to ASTM/ASME fabrication and material specifications. This ensures that the elastic analysis methods of ASME BPVC Section III, including stress intensity factors are valid and offer a high degree of conservatism.

The surveillance test program specimens are used to monitor the radiation-induced changes in the mechanical properties of the core beltline region materials of the RPV. Part of the specimens that are exposed to irradiation are installed in removable specimen capsules at the inside vessel wall opposite the active core. The remaining unirradiated baseline specimens establish material are used to establish reference data.

The impact of irradiation on ferritic pressure boundary materials is described in ASME Section III, Non-mandatory Appendix W, Paragraph W-3300. As part of the design, vessel irradiation (neutron irradiation) is calculated over the 60-year design life of the plant. Because the core is the source of the neutrons, the fluence will be at a maximum in the region surrounding the active core, which is consistent with all previous BWR designs.

Surveillance specimen materials are prepared from samples are taken from the actual materials used to fabricate the beltline of the RPV. The weld and Heat Affected Zone (HAZ) samples are supplied if they are exposed to neutron fluence  $1.0E17$  n/cm<sup>2</sup> over the design life of 60 years.

The predicted changes from the initial properties are a function of chemical composition and the neutron fluence during reactor operation. The base and weld material with the highest predicted adjusted reference temperature at end of life are selected for the surveillance program. The prediction of adjusted reference temperature shift is in accordance with peak neutron fluence value specified in the vessel irradiation data drawing.

### **5.2.3 Overview of Reactor Pressure Vessel and Pressure Boundary Components**

The RPV and associated core support structures and piping are designed to:

- Withstand static and dynamic loading, including thermal expansion and contraction
- Withstand vibration (such as flow induced and acoustic vibration)
- Ensure chemical compatibility, including service-related contaminants
- Meet thermal material limits
- Meet radiation damage limits

The primary materials used in the RPV and pressure boundary components are listed in Table 5.2-1, together with the applicable specifications. The RPV materials comply with the provisions of ASME BPVC Section III and shall also meet the requirements of ASME BPVC “Section II Materials” (Reference 5.2-1). The RPV materials also meet the additional requirements to address potential degradation mechanisms. These materials provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful operating experience in reactor service.

#### **5.2.3.1 Reactor Pressure Vessel and Nozzles**

The RPV is constructed from low alloy, high strength steel forgings or plates. Forgings are ordered to ASME material specification SA-508M, Grade 3, Class 1, while plates are ordered to ASME SA-533M, Type B, Class 1. This material is melted to fine grain practice and is supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Table 5.2-1 lists the specifications for the RPV Vessel and Nozzle materials. Nominal compositions for these low alloy steel and carbon steel materials used in reactor components are presented in Table 5.2-2. The details of all smaller components such as low alloy steel studs, nuts, and other components are consistent with the material in Table 5.2-1. Welding electrodes for low alloy steel are low hydrogen type ordered to ASME SFA-5.5, and weld filler metal to SFA-5.23 and SFA-5.28. All forgings and bolting are 100percent ultrasonically tested and surface examined as required by ASME BPVC Section III, Division 1 (Reference 5.2-2). Fracture toughness properties of materials are also measured and controlled in accordance with ASME BPVC Section III, Division 1 as discussed in Subsection 5.2.4.

#### **5.2.3.2 Alloy Steel, Stainless Steel and Carbon Steel Piping**

The information for the main components provided in Tables 5.2-1 and 5.2-2 include the specifications and typical chemistry of stainless steel, low alloy steel and carbon steel piping components. These piping components are processed to achieve a fine grain, microstructure.

#### **5.2.3.3 Reactor Pressure Vessel Weld Materials**

Weld materials used for the RPV construction as well as the RPV nozzle attachment processes are contained in Tables 5.2-1 and 5.2-2. The ASME/ASTM specifications are listed that provide the details for the type of weld material, covered electrode or filler metal, that are used in making the weld attachments.



#### **5.2.3.4 High Strength Reactor Pressure Vessel Bolting Materials and Fasteners**

The high strength low alloy bolting materials used in the RPV closure or for flange application are generally Type 4140 or Type 4340 high strength steels. The typical specifications and compositions are listed in Tables 5.2-1 and 5.2-2.

The maximum measured yield strength and surface hardness of the stud bolting materials (studs, nuts, and washers) shall meet ASME Code requirements. Each stud is tested after rough machining on the outside surface and the center of an end position. Hardness tests is performed on all main vessel closure bolting to demonstrate that heat treatment has been properly performed. Given their importance, studs, nuts, and washers are individually tested to assure they meet specification limits.

#### **5.2.4 Degradation Mechanisms for Reactor Pressure Vessel and Pressure Boundary Components**

Component degradation mechanisms that are applicable to both the steel vessel and piping materials are assessed and controlled to assure component integrity. Degradation mechanisms include radiation embrittlement, general corrosion, and FAC. RPV fracture toughness can be affected by the operating environment; therefore, fracture toughness requirements are specified.

##### **5.2.4.1 Reactor Pressure Vessel Toughness Requirements**

ASME Section III requirements apply to all pressure boundary materials. Specific toughness values at the Reference Nil-ductility Transition Temperature ( $RT_{NDT}$ ) are determined to assure the RPV maintains adequate fracture resistance at the lowest use temperatures. The different product forms have specific requirements that are evaluated, including the beltline shell courses, nozzle forgings, safe ends, and weld filler metal.

To assure that the steel components have adequate toughness, impurity limits are set on Cu, P and S for the base metal and Cu, P, V and S limits for the weld filler metal.

##### **5.2.4.2 Radiation Embrittlement (Vessel Beltline) and Loss of Toughness**

For all material exposed to a neutron fluence of more than  $1 \times 10^{17}$  n/cm<sup>2</sup> for which impact tests are required, fracture toughness ( $K_{IC}$ ) shall be established as a function of temperature over the range of  $RT_{NDT}$  to  $RT_{NDT} + 33^{\circ}\text{C}$ . Testing requires the use of specimens with adequate thickness with details given in ASME BPVC, "Section II Materials" (Reference 5.2-1) and Subsection 5.11.8.2 entitled RPV Materials Surveillance Programs.

##### **5.2.4.3 General Corrosion**

The BWRX-300 addresses time dependent corrosion, other than localized corrosion degradation, such as IGSCC, TransGranular Stress Corrosion Cracking and Irradiation Assisted Stress Corrosion Cracking (IASCC). For various operating conditions, corrosion allowances are incorporated into the design process. These allowances need to account for the impact of the Hydrogen Water Chemistry operating environment as well as the flow rates associated with standard operation (Hydrogen Water Chemistry might affect these rates). During design, general corrosion allowances for operation and shutdown are also considered. Alternate design allowances for specific components or systems are applied based on the local environment and operating conditions. To determine the total corrosion allowance, appropriate operating allowances for all plant conditions are added together to develop the overall required allowance.

##### **5.2.4.4 Flow Accelerated Corrosion**

General corrosion for the plant components is managed in the design process based on the ASME Code as stated in the previous section. However, under certain conditions of water chemistry,

system flow, geometry, and component material, FAC can degrade piping and components made of carbon steel (with low chromium composition). The typical compositions for the carbon steel and alloy steel that are used for piping and piping components are given in Table 5.2-2. Options to ensure mitigation against FAC include increasing piping diameter, increasing elbow radius, using higher alloyed steels, or using stainless steel.

For piping made of carbon steel, extensive research has been performed to avoid FAC. The key factors that can affect the FAC rate are: (1) Temperature; (2) Material Composition; (3) Flow rate; (4) pH of the water; and (5) oxygen content. Each of these factors is discussed below:

- Temperature – generally in the range of 100°C to 150°C (can also occur at higher temperatures, but becomes less significant)
- Material Composition – materials containing <0.25 wt percent of chromium
- Flow Rate – Fully developed turbulent flow (example: Reynolds number > 150,000)
- pH – environments with a pH <8
- Dissolved Oxygen – Concentrations < 30 ppb

These parameters provide the necessary understanding to allow the design team to assure that the concerns for any carbon steel lines are managed for the BWRX-300.

#### **5.2.4.5 Additional Time Sensitive Degradation**

Other than the risks of SCC, material data is required to address the effects of thermal conditions that could change the microstructure thereby reducing the ductility and toughness of the materials. This is applicable to the ferrite levels in cast stainless steel materials as controlled by the specifications USNRC NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" (Reference 5.2-3). As discussed, FAC is a specific concern for carbon steels. Additionally, there are other types of loading that can affect material integrity. Specifically, cyclic loading, commonly referred to as fatigue, occurs in different regions and components in the BWR. These are addressed using ASME Code design rules.

#### **5.2.4.6 Bolting Relaxation**

Concerns for fastener relaxation are addressed by the initial material selection process, the design process that sets the applied stresses to assure elastic loading and the installation sequence and loading process. The design process also addresses any relaxation attributable to the operating environment. These guidelines are also complemented with the use of continual review of fastener performance in current operating BWRs.

### **5.2.5 Overview of Core Structural Component Materials**

It is important that the materials to be used in the BWRX-300 are capable of maintaining reliable operation of plant systems and equipment during the design life of each component. For the reactor internals, austenitic stainless steel is primarily used to minimize levels of radiation from corrosion products generated in the core. Where higher strength requirements than those for standard austenitic stainless steels are needed, nickel base alloys, nitrogen strengthened austenitic alloys, or precipitation hardening stainless steels are used.

Materials selection with composition controls and defined fabrication processes is used to address materials degradation issues in the reactor system specifically SCC and irradiation SCC. Figure 5.2-1 shows the three factors that lead to degradation. Selection of the materials is directed toward minimizing their susceptibility. Design guidelines and controlled fabrication processes are used to manage the component stresses, with emphasis on surface stresses. It should be noted that the impact of the coolant environment is addressed using Hydrogen Water Chemistry/OLNC.

The stainless steel structural materials used in the BWRX-300 reactor core internals are primarily wrought materials along with their weld metals although Cast Austenitic Stainless Steels are used as well. It is noted that no Cast Austenitic Stainless Steels will be used for the reactor pressure vessel. However, Cast Austenitic Stainless Steels may be utilized in other pressure-retaining, cyclic, and/or dynamic load applications when supported by appropriate design analysis evaluations.

With respect to reactor internals and/or core support applications (see Subsection 5.2.5), Cast Austenitic Stainless Steels may be utilized. As an example, fuel supports are subject to some cyclic and dynamic loads during reactor operation and operating events. Cast Austenitic Stainless Steels is used for the fuel support due to the complexity of the part and challenges fabricating from wrought material. Appropriate chemistry controls to mitigate material degradation concerns will be applied. Operating experience with this material over the past 60 years supports its use in this application.

Table 5.2-3 lists the materials used for the typical internal components including the shroud, chimney, steam dryer, top guide, core support, CRD housings and instrumentation housings. The table lists the material product form as well as representative ASME/ASTM specifications. Table 5.2-4 lists typical compositions for these austenitic reactor component materials.

#### **5.2.5.1 Wrought Austenitic Stainless Steels**

The primary alloys to be used for reactor internal components are austenitic structural alloys. Specifically, the wrought austenitic stainless steel components are Type 304, 304L, 316, 316L and potentially 316L with higher nitrogen content.

These materials have a long history of use in all past BWR applications. For L grade materials, the primary controlled constituent is carbon. Carbon is restricted to 0.03 weight percent to provide resistance to sensitization during welding while still being able to meet the specified strength levels at the use temperature. For applications in high fluence regions of the core, residual elements are controlled to reduce the susceptibility of the steel. The efforts to maintain higher strength at temperature led to the development of special versions of Type 316L (originally designated Nuclear Grade which include nitrogen additions to maintain high temperature strength in these low carbon alloys). The chemistry of Type 316/316L material for components is also controlled to have positive ferrite potential as determined by ASTM A800/A800M-20, "Standard Practice for Estimating Ferrite Content of Stainless Steel Castings Containing Both Ferrite and Austenite" (Reference 5.2-4) or the use of an approved alternate ferrite diagram for composition control with a target level of 3 percent. This is necessary to assure good weldability. The additional controls on alloy composition are also given in notes to Table 5.2-3. These austenitic materials are given a solution heat treatment at a temperature range with a hold time designed to achieve a full solution annealed condition while minimizing grain growth. The heat treatment is then followed by accelerated cooling to avoid any sensitization. It should be noted that Alloy XM-19 with its higher strength, developed for control rod drive components, will be used in special locations due to its excellent IGSCC resistance.

#### **5.2.5.2 Cast Austenitic Stainless Steel**

These cast versions of the wrought materials rely on carbon control and the presence of ferrite to assure IGSCC resistance which requires the ferrite level of 8 percent minimum for SCC. Additionally, the ferrite levels are also restricted to less than 20 percent to assure adequate toughness after long-term exposure to higher temperatures. The castings are given solution annealing to maintain IGSCC resistance. Prior operating experience has validated resistance to IGSCC (ASTM STP 756, "Intergranular Stress-Corrosion Cracking Resistance of Austenitic Stainless Steel Castings" (Reference 5.2-5)).

### **5.2.5.3 Nickel Base Alloys**

Similar composition controls and product form processing to stainless steels are used to assure that wrought Ni-Cr-Fe Alloy 600 (including modified Alloy 600) and the associated Weld Metal Alloy 82 (ASME BPVC, "Case N-580-2 Use of Alloy 600 with Columbium Added Section III, Division 1" (Reference 5.2-6)) are free of any sensitization risks. Nickel base wrought materials are Alloy 600 with niobium modified chemistry. Specifically, the materials need to have a stabilizing ratio  $(Nb + 2 \cdot Ti/C) > 92$ . Additionally, nickel base filler metals are to be Alloy 82 (ERNiCr-3) with a stabilizing ratio that also follows the same requirements.

### **5.2.5.4 High Strength Alloys**

In special locations, higher strength Alloy X-750 is used for bolting and fastener type applications. The main approach used to prevent SCC susceptibility is accomplished by limiting the applied stresses in the design process. High strength austenitic Alloy X-750 is procured and processed to assure adequate SCC resistance. Additionally, Alloy 718 is considered to be used for high strength applications. Alloy 17-4PH is a martensitic precipitation hardening stainless steel of high strength, therefore, potentially susceptible to SCC if the hardness is not limited.

## **5.2.6 Overview of Design and Fabrication Process for SCC Controls in Reactor Core Components**

To prevent SCC, the third factor that is necessary is the presence of elevated surface stresses. These residual stresses are often introduced by fabrication such as welding, surface grinding or forming operations. These types of stresses will support crack initiation on the wetted surface. For austenitic stainless steels or nickel base alloys, construction process controls are applied during various stages of component manufacturing and reactor construction to avoid these fabrication-induced surface stresses that could lead to SCC initiation.

These processing steps can also introduce localized sensitization. The processes that need to be controlled include bending or forming, final machining, grinding, polishing, and welding. Each type of operation and an overview of the controls to be applied are summarized in more detail below.

### **5.2.6.1 Forming Processes**

These processes are required for piping or cylindrical structures. It is important that forming is controlled and does not introduce significant strains that could lead to SCC initiation if the shape is not given a post forming solution anneal. These processes are limited to 2.5 percent total strain to prevent future SCC. The GEH materials and process controls define these limits.

### **5.2.6.2 Machining, Grinding and Polishing**

Surface machining, and grinding are required as part of large component fabrication. Specific procedures employing specific media and subsequent polishing using increasingly finer media and decreasing material removal are also specified. These polishing procedures are particularly used following welding in the fabrication of reactor internal components such as the core shroud. Process qualification always precedes actual practice. The primary regions of potential susceptibility are the welds and their HAZ regions. Specifically, abrasive grinding of stainless steel HAZ regions is minimized to the extent possible. When grinding is applied to the weld regions, care is taken to confine grinding to the weld metal region only and limit grinding of the adjacent base metal. This is necessary to meet the required fabrication and examination requirements of ASME. Grinding abrasives are controlled to avoid detrimental effects on austenitic materials. Grinding is followed by polishing processes to further reduce the risks of SCC.

The use of the final polishing processes is restricted to those with qualified procedures. The procedure contains methods that involve finishing the ground surface with successively finer grit size such that the bulk of any surface cold work is removed. Finishing with each grit size continues until evidence of the previous grit is removed.

#### **5.2.6.3 Welding Processes**

Over the history of BWR plant construction and operation, GEH has refined the technologies for fabrication, maintenance, and repair of welded components in nuclear reactors. Each process improvement has increased margins against SCC in BWRs by controlling residual stresses, alloy chemistry, and microstructure for the welds and surrounding HAZ regions. GEH developed specialized narrow groove welding technology, which is an automated welding technology addressing these three critical factors. Specifically, it produces compressive stresses in the weld root regions. The narrow weld joint design improves fabrication productivity and reduces weld shrinkage distortion. The low heat input, combined with special welding parameters, also limits HAZ residual strains and sensitization in addition to the compressive stresses, which will further prevent SCC. Since its development, the process has been used across the power generation industry.

#### **5.2.6.4 Electro-Discharge Machining**

Under special circumstances, Electro-Discharge Machining (EDM) is used. Specific controls, developed in previous BWR applications, are used on nickel base alloys to prevent surface degradation such as microfissuring.

In summary, GEH uses developed fabrication processes with a long history of refinement to prevent SCC initiation in components made using austenitic structural alloys. These fabrication processes are also used to prevent IASCC and TransGranular Stress Corrosion Cracking.

#### **5.2.6.5 Additional IASCC Material Controls**

As with SCC, the parameters that contribute to IASCC in a BWR environment are material selection with its processing, the applied and residual stresses in the components, and the magnitude of the neutron radiation exposure. Type 316L is the alloy of choice for the core region.

Table 5.2-4 includes a listing of special chemistry controls on silicon for Type 316L when used for reactor internals subjected to IASCC. The benefits of the control of silicon are confirmed by research and are used in core components, specifically the core shroud.

#### **5.2.6.6 Specific TransGranular Stress Corrosion Cracking Fabrication Controls**

IGSCC and TransGranular Stress Corrosion Cracking often occur in the same alloys depending on the environment or the microstructure. However, cases of TransGranular Stress Corrosion Cracking in small lines and other components, have been associated with contaminants such as chlorides that can be present on the material surface.

These contaminants, related to salts or marking materials, are also controlled during fabrication as well as operation. Miscellaneous non-metallic materials, used for the fabrication and assembly of reactor system components, are also controlled to minimize contamination with species known to have detrimental effects on alloys such as stainless steels and nickel alloys.

Those materials which come in contact with stainless steel or nickel alloys during these processes are either low in halogens, sulfur, and lead or other low melting alloys or need to be removed in the final cleaning of the part or assembly. The known contaminants of concern are chlorine, fluorine, sulfur, lead, mercury, and all other low melting point metals. In addition, when welding or solution heat treatment is involved, phosphates and carbonaceous materials are to be removed prior to welding or heat treatment.

#### **5.2.6.7 Sensitivity to Fracture**

The alloys used in BWR systems other than the RPV and steel components are predominantly alloys with lower strength and high ductility. These alloys are not susceptible to rapid fracture as substantiated by the excellent operating history of BWR components fabricated from austenitic stainless steel and nickel base alloys. The only components that show loss of ductility and reduced fracture resistance are the core components, particularly the core shroud, which is subjected to irradiation. The irradiation leads to material changes, hardening and grain boundary segregation, that in, turn, increase the strength significantly and reduces the ductility. The same factors that increase the risk of IASCC, also reduces fracture resistance. The shroud design takes these factors into account to assure adequate robustness. The design also positions the shroud welds to reduce the fluence experienced by the weld region. Cast Austenitic Stainless Steels material compositions restrict the ferrite level to assure toughness is not lost with thermal aging.

#### **5.2.7 Gasket, Seal, and Fastener Materials**

##### **5.2.7.1 Gaskets and Seals**

These non-metallic engineered materials used in association with reactor system components are controlled to minimize or eliminate potential for detrimental effects on metallic reactor components. These materials which include gaskets, packing, bushings, etc., that are installed within the reactor system such that they are in contact with reactor water conform to the chemistry controls of specifications. Examples of the specific materials used include Viton, Buna-N, EPDM and graphite non-metallic seals as well as metallic seals including silver plated Alloy 718. The specifications set limits on halogens, sulfur, and nitrates in the non-metallic materials. There are also specific prohibitions or limitations on the use of chlorine and fluorine bearing materials such as Teflon and polyvinyl chloride. Exceptions are allowed but must be evaluated on an individual basis considering such factors as location, radiation dose, temperature, and successful operating history.

##### **5.2.7.2 Fasteners**

Sections 5.2.3 and 5.2.5 include discussion of low alloy steel and austenitic high strength materials, respectively, that are used as fasteners. There is extensive experience in operating BWRs with these materials that are used as the basis for fastener selection for use in the BWRX-300. Tables 5.2-1 through 5.2-4 include the specifications and compositions of these special use materials.

#### **5.2.8 References**

- 5.2-1 ASME BPVC, "Section II Materials," American Society of Mechanical Engineers.
- 5.2-2 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 5.2-3 USNRC NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."
- 5.2-4 ASTM A800/A800M-20, "Standard Practice for Estimating Ferrite Content of Stainless Steel Castings Containing Both Ferrite and Austenite," American Society for Testing and Materials.
- 5.2-5 ASTM STP 756, "Intergranular Stress-Corrosion Cracking Resistance of Austenitic Stainless Steel Castings," American Society for Testing and Materials.
- 5.2-6 ASME BPVC, "Case N-580-2 Use of Alloy 600 with Columbium Added Section III, Division 1," American Society of Mechanical Engineers.

**Table 5.2-1: Typical Materials for Reactor Pressure Vessel Components**

<b>Components</b>	<b>Form</b>	<b>Material Type</b>	<b>Typical ASME/ASTM Specification</b>
RPV Shells and Heads	Plate	Mn-1/2Mo-1/2 Ni Low Alloy Steel	SA-533M Type B, Class 1
	Forging	3/4 Ni-1/2Cr-Cr-V Low Alloy Steel	SA-508M Grade 3, Class 1
RPV Nozzle Safe Ends	Forging	Low Alloy Steel	SA-508M Grade 3, Class 1
RPV Main Closure Bolting, Flange Bolting	Bolting	Low Alloy Steel	SA-540M Grade B23 or B24, Class 3
Low Alloy Steel Piping	Seamless Forged & Bored	Low Alloy Steel	SA-336M Grade P22, SA-369M FP22
Carbon Steel Piping	Seamless	Carbon Steel	SA-333M Grade 6
Carbon Steel Welds	Covered Electrode or Filler metal	Carbon Steel	SFA-5.1 SFA-5.18
Low Alloy Steel Welds	Covered Electrode or Filler metal	Low Alloy Steel	SFA-5.5 SFA-5.23 SFA-5.28
Low Alloy Steel Piping Welds (2-1/4 Cr 1 Mo)	Covered Electrode or Filler metal	Low Alloy Steel (2-1/4 Cr 1 Mo)	SFA-5.5 SFA-5.28
Stainless Steel Welds	Covered Electrode or Filler metal	Stainless Steel	SFA-5.4 SFA-5.9
Nickel Alloy Welds	Filler Wire	Nickel Alloy	SFA-5.14

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 5.2-2: Typical Compositions of Low Alloy Steel and Carbon Steel Materials**

	<b>C</b>	<b>Cr</b>	<b>Mn</b>	<b>Mo</b>	<b>Ni</b>	<b>Fe</b>	<b>P</b>	<b>S</b>	<b>Si</b>	<b>Other</b>
SA-533M, Gr. B (Note 1)	.25 max		1.15- 1.5	.45 - .60	.40 - .70	Rem	.025 max	.025 max	.15 - .40	
SA-508M, Grade 3 (Note 1)	.25 max	1.50 – 2.00	1.20 - 1.50	.45 - .60	0.50- 1.00	Rem	.025 max	.025 max	.40 max	Restricted V, Cu, Nb, Ca, Cu, B
SA-540M Grade B23 (Note 2)	.37- .44	.65- .95	.6-.95	.2-.3	1.55- 2.00	Rem	.025 max	.025 max	.15- .35	
SA-333M Grade 6 Carbon Steel (Note 3)	0.30 max	0.3 max	0.29- 1.06	0.12	0.4	Bal.	0.025 max	0.025 max	0.10 (min)	-
SA-335M Grade P22 (2 ¼ Cr-1 Mo)	0.5- 0.15	1.9- 2.6	0.3- 0.6	0.87- 1.13	-	Bal.	0.025 max	0.025 max	0.5 max	-

- (1) There are special controls of key elements and interstitials to address different embrittlement processes including radiation embrittlement and reduction of toughness in the RPV.
- (2) Maximum allowable yield strength 1034 MPa (150 ksi).
- (3) There are also composition controls to address Flow Accelerated Corrosion (FAC) that are applicable to standard carbon steel piping. These mechanisms are discussed in the next sections.



**Table 5.2-3: Typical Reactor Internal Components and Materials**

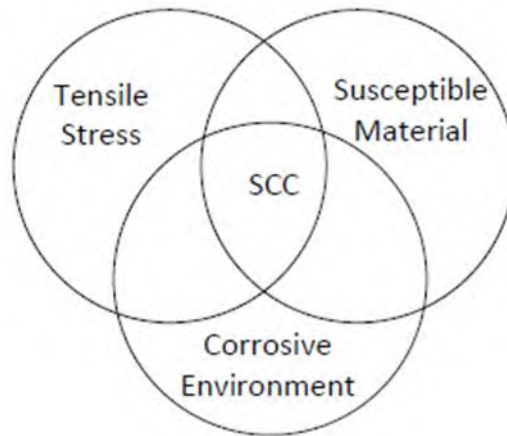
<b>Component(s)</b>	<b>Form</b>	<b>Material Type</b>	<b>Typical ASME/ASTM Specification</b>
Steam Dryer	Plate, Bar or Forgings	Stainless Steel	SA-240M, SA-182M, SA-479M Type 304/304L, 316/316L
Core Shroud and Support Ring	Plate or Forgings	Stainless Steel	SA-240M, SA-182M Type 316L
Core Support Structure	Plate or Forgings	Stainless Steel	SA-240M, SA-182M Type 316L
Core Support Legs	Plate, Bar or Forgings	Columbium modified Ni-Cr-Fe (A600M) Nickel Base Alloy	SB-168, SB-166 or SB-564 Nickel Alloy modified per ASME Code Case N-580-2
Top Guide	Plate or Forgings	Stainless Steel	SA-240M, SA-182M Type 304/304L, 316/316L
Chimney	Plate or Forgings	Stainless Steel	SA-240M, SA-182M Type 304/304L, 316/316L
Control Rod Drive Housings	Seamless Pipe or Forgings	Stainless Steel	SA-312M, SA-182M Type 304/304L, 316/316L
Fuel Supports	Castings	Stainless Steel	SA-351M Grade CF3
In-core Housings	Seamless Pipe or Forgings	Stainless Steel	SA-312M, SA-182M Type 304/304L, 316/316L
Control Rod Drive Penetration Stub Tubes	Bar, Pipe or Forging	Ni-Cr-Fe Alloy	SB-166, SB-167 or SB-564 Nickel Alloy 600 modified per Code Case N-580-2
Core Plate Bolts	Bar	XM-19	SA-479M Type XM-19
Stainless Steel Welds	Covered Electrode or Filler metal	Stainless Steel	SFA-5.4 SFA-5.9
Nickel Alloy Welds	Filler Wire	Nickel Alloy	SFA-5.14

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 5.2-4: Typical Compositions: Standard BWR-300 Materials and Special Compositional Controls for Toughness and IASCC Resistance**

	<b>C</b>	<b>Cr</b>	<b>Mn</b>	<b>Mo</b>	<b>Ni</b>	<b>Fe</b>	<b>P</b>	<b>S</b>	<b>Si</b>	<b>Other</b>
316L	0.030	16-18	2.00	2-3	10-14	Bal.	0.045	0.030	0.75	-
304L	0.030	17.5-19.5	2.00	-	8-12	Bal.	0.045	0.030	0.75	-
316NG (316L plus N)	0.020	16-18	1.0-1.5	2-3	12-14	Bal.	0.025	0.01	.25	N (0.06-0.10)
Alloy 600	0.15	14-17	1.0	-	72 (min)	6-10	-	0.015	0.5	Cu 0.5.
Alloy 600M*	0.050	14-17	1.0	-	72 (min)	6-10	-	0.015	0.5	Cu 0.5 Nb 1-3
Alloy X-750	0.08	14-17	1.00	0.7-1.2	70 min.	5-9	-	0.01	0.5	Ti 2.25-2.75 Al 0.4-1.0
Alloy 718**	0.08	17-21	0.035	2.8-3.3	50-55	Bal	0.015	0.015	0.035	Nb, Ti, Al, Co
XM-19	0.04	20.5-23.5	4-6	1.5-3	11.5-13.5	Bal.	0.045	0.030	1.00	Nb 0.1-0.3; V 0.1-0.3
Alloy 17-4PH	0.07	15-17.5	1.00	-	3-5	Bal.	0.040	0.030	1.00	Cu 3-5

- (1) \* Nb-modified Alloy 600 per BPVC Case N-580-2.
- (2) \*\*Additional constituents: Nb: 4.75-5.5; Ti: 0.65-1.15; Al: 0.2-0.8; Co: 1.0 max.
- (3) All single values are maximums, unless otherwise specified.
- (4) Controlled Residuals for IASCC Resistance - Type 316L: Silicon controls: Silicon in the core shroud and top guide shall be limited to 0.10 percent maximum.
- (5) Controlled Composition for Improved Weldability - Stainless Steel: Composition controls to keep ferrite levels >3%.



**Figure 5.2-1: The Three Necessary Factors to Produce Stress Corrosion Cracking**

### 5.3 Nuclear Boiler System

The NBS is comprised of three primary subsystems: RPV, MS, and RPV instrumentation. In addition to these subsystems, the NBS also provides RPV head venting and RPV head flange seal leak detection features. Within the NBS are the redundant with diverse instrument and electrical features, safety class reactor isolation valves interfacing with the subsystem and systems listed below:

- MS (Subsection 5.7.1.4 and Chapter 10, Section 10.4)
- RPV head vent (Subsection 5.12.3)
- ICS steam supply (Section 5.7 and Chapter 6, Section 6.2)
- ICS condensate return (Section 5.7, Chapters 6 and 9)
- Condensate and Feedwater Heating System (CFS) (Chapter 10, Subsection 10.3.2)
- CUW (Subsection 5.12.2 and Chapter 9A, Subsection 9A.2.2)

#### 5.3.1 Nuclear Boiler System Configuration

The NBS (see Figure 5.1-2) consists of the RPV, two main steam lines, MSRIVs, FWRIVs, IC Supply RIVs (ICSRIV), IC Return RIVs (ICRRIV), CUWRIVs, MSCIVs, steam line drain subsystem, RPV Head Vent RIVs and Head Vent subsystem, a Head Flange Leak Detection subsystem, nuclear instrumentation, and reactor pressure, reactor water level, steam, temperature, and leak detection instrumentation.

The CFS provides high purity feedwater to the RPV at the required flow rate, pressure, and temperature. The CFS interfaces with the NBS just before injection into the reactor vessel located in the Reactor Building. The CFS receives main steam from the NBS for high-pressure Feedwater Heater (FWH) number 6. Only the portion of the CFS from the CFS CIVs to the outboard FWRIVs outboard flange to is part of the RCPB as indicated in Section 5.1, and it is described further in Chapter 9A.

##### 5.3.1.1 Reactor Pressure Vessel Description

The NBS utilizes an RPV to house and support the core and internals, provide a pressure boundary and flow path for reactor coolant, interface with CFS as the coolant supply, generate steam for the main turbine, and provide emergency core cooling via the ICS. The RPV nozzles greater than DN20 are flanged to two integral in-series RIVs. The nozzles are for the MS (N1), ICS Supply (N2), FW (N3), ICS Return (N4), CUW (N5), and Head Vent (N6) systems or subsystems. Nozzles with a bore less than DN20 are not required to have integral in-series RIVs. The DN20 and below nozzles are for Level Low-Pressure or reference leg (N7), Level High-Pressure or variable leg (N8), and Head Flange Leak Detection (N9).

There is one nozzle (N10) greater than DN20 that does not have RIVs and is located at least TAF+4m. The purpose of the N10 nozzle is for required prototype testing that supports the internal vibration monitoring. That nozzle would normally have a blind flange installed. When the internal vibration program is in operation, another flange is installed that has individual holes for the instrument cabling installed through interfacing bolts.

##### 5.3.1.2 Main Steam Subsystem

The MS subsystem is comprised of two main steam lines that are routed from the outboard MSRIV discharge through the outboard MSCIV out to the TSVs, associated branch piping, and the MSL drains.

The MS piping (in Containment and RB) from the discharge of the outboard MSRV to the Seismic Interface Restraint (SIR) is designed to ASME BPVC Section III, Division 1, Class 2, Seismic Category A requirements following USNRC NUREG-0800 BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" (Reference 5.3-1) guidance. The MS piping from the outboard MSRV discharge to the low point of the piping just inside the TB is designed to have a negative slope such that any steam condensate will drain outwards to the MS drain lines.

The MS piping in the TB, including all the connecting branch lines up to the various steam loads and their respective isolation valves (see Chapter 10) is designed to ASME B31.1, "Power Piping" (Reference 5.3-2), which has non-seismic category requirements. The MS equalizing header supplies high-pressure steam via branch piping to the TBVs, the moisture separator reheater (MSR), the number 6 Feedwater Heater (FWH), and the Turbine Auxiliary Steam Subsystem (TASS).

The MS pipeline routing includes sufficient length such that the MSL minimum volume requirements are met using the pipe sizes designated from the system Piping and Instrumentation Diagram (P&ID) and Process Flow Diagram (PFD).

The MS subsystem is designed so that the pressure drop through the piping at 100 percent Rated Thermal Power (RTP) condition does not exceed the minimum or maximum limits.

MSLs are designed to minimize the potential for water (steam) hammer through the implementation of specific design features and system layout(s). The design also complies with USNRC NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants" (Reference 5.3-3) as applicable.

The BWRX-300 design includes an outboard MSCIV (see Figure 5.1-3) on each MSL. The MSCIVs provide SC1 containment isolation for accidents or events to prevent unfiltered release of radiological contaminants through the MSLs that exceed limits. This containment isolation is single failure proof. The MSCIVs are fast-closing and fail-closed type valves. These isolation valves outside of containment are located as close to the containment as practical to satisfy the containment isolation design requirements.

The MSCIVs are designed in accordance with the rules and requirements of ASME BPVC, Section III, Division 1, Subsection NCD, Class 2 and Class 3 Components (Reference 5.3-4), in accordance with their quality group classification. The MSCIVs are designed for Seismic Category B and certified by a qualification program prepared and performed in accordance with ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities" (Reference 5.3-5).

The MS drain lines provide the ability to drain the condensate from the MS lines to the condenser in a controlled manner during startup, lower power operation (approximately 40 percent RTP for drain valves control), normal power operation, and shutdown. The MSL drain isolation valves incorporate control logic interlocked to support automatic plant startup and shutdown operation.

### **5.3.1.3 Nuclear Boiler System Instrumentation Subsystem**

#### **5.3.1.3.1 Reactor Water Level**

The water level in the reactor vessel is measured by the temperature compensated dP devices (calibrated for specific RPV pressure and temperature conditions).

Historically, BWR water level has been monitored over both NR and wide range (WR). Previous RPV designs have multiple taps to cover the different ranges. The NR is calibrated for a smaller range of elevations and therefore can be held to a higher accuracy than WR. This is because the

measurement accuracy is linearly related to the overall differential height of the sensor pair. During normal operations, NR is used to control the target water level and FW flow. WR level monitoring is used during off-normal events. In the case of BWRX-300, the added safety features of not having RPV penetrations below TAF + 4m, as well as minimizing the number of nozzles, results in having only one set of nozzles for each division of water level monitoring. Therefore, NR and WR for each division are taken from the same set of nozzles.

The reactor water level measuring system is classified SC1 and is monitored by the C10 Distributed Control and Information System (DCIS). The C20 DCIS is also used to monitor the water level for both SC2 and SC3 functions. The BWRX-300 is designed to have three divisions for monitoring and control of SC1 functions, as well as redundancy within each division (refer to Chapter 7 for detailed description of control logic architecture).

Another key function of the water level measurement system is to provide shutdown range water level monitoring as the RPV is flooded up before the head is removed. It will also monitor water level during refueling and support post-outage restart. This is a SC3 function with two pressure transducer pairs.

Water level measurement is not required between TAF and TAF + 4m because there are no operator actions associated with this range. In the case of a BDBAs and DECAs where monitoring water level in the core (also known as fuel zone range) is necessary, it will be accomplished with GTs. The GTs are configured with a heater wire to provide a heated junction to every location on a string used to support water level monitoring even when gamma flux is low, and also to provide testing and calibration capability.

#### **5.3.1.3.2 Reactor Pressure Instrumentation**

RPV steam dome pressure is measured using wide range and narrow range differential pressure instruments to provide pressure indication in Modes 1 through 5. The RPV steam dome pressure measurement supports Post-Accident Monitoring (PAM) system.

#### **5.3.1.3.3 Core Flow Instrumentation**

Mass core flow in the BWRX-300 RPV is not measured directly but is calculated from an aggregation of temperature measurements throughout the core and RPV pressure. The core flow is calculated by the heat balance core flow methodology using the core inlet temperature measurement as input to determine core inlet enthalpy. The methodology provides a verified basis for calculated core flow.

#### **5.3.1.3.4 Main Steam Flow Instrumentation**

The main steam flow restrictors are located in the RPV MS nozzles. They perform functions in various safety classifications and corresponding defence levels. The SC1 functions are to limit the overall steam flow through one steam line when the steam flow exceeds preselected operational limits (i.e., if a large break of a MSL is sensed). The other SC1 function is to provide that same sensing function to the SC1 MSRIV/MSCIV isolation system. The SC2 steam flow function also supplies the same sensing functions for MSRIV/MSCIV isolation in addition to the already discussed SC1 isolation function. The SC3 steam flow function supports the overall reactor water level controller that maintains target water level as one of the inputs to that controller.

The main steam flow restrictor provides low-pressure sensing taps for instrumentation lines of differential pressure transmitters that are used to measure steam flow. Main steam flow is measured through the RPV nozzle flow restrictors using Bernoulli's equation of differential pressure to derive the MS mass flow. The mass flow can then be calculated by knowing the steam properties based on RPV pressure and temperature.

#### **5.3.1.3.5 Reactor Power Instrumentation**

Reactor power instrumentation is described in Subsection 5.4.2.

#### **5.3.1.3.6 RPV Metal Temperature Instrumentation**

The RPV outside surface (metal) temperatures are measured at the head flange and the bottom head locations in order to comply with plant Operational Limits and Conditions (OLC) (Technical Specification) surveillance RPV heat-up and cooldown requirement limits which define allowable operating regions and permits a large number of operating cycles while also providing a wide margin to nonductile failure.

### **5.3.2 NBS Classification of SSC**

The overall NBS is classified as a SC1 system following the methodology for classification of BWRX-300 Structures, Systems and Components (SSCs) as discussed in Chapter 3, Section 3.2. Table 3A.5-1 provides a list of the primary BWRX-300 SSCs and their classification in accordance with safety class, seismic category, and quality group.

### **5.3.3 NBS Functions**

The NBS components are designed, specified, procured, fabricated, tested, installed, and commissioned in accordance with the codes and standards specified in Chapter 1, Appendix B. Chapter 1, Appendix B lists the approved codes and standards and identifies the applicable editions used for the design of mechanical systems and components of the BWRX-300 including those described in Chapter 5. The pressure boundary codes and standards are listed in Table 3.6-1 of Chapter 3, Section 3.6.

#### **5.3.3.1 Normal Functions (Non-safety Category)**

The NBS is designed to fulfill the following normal and off-normal safety functions. The normal operating modes including description of how the plant operates in these different operating modes is provided in Chapter 16, Appendix 16A.

- The BWRX-300 RPV operates through natural recirculation
- NBS converts heat energy resulting from the fission process in nuclear fuel into heat energy for electric power production in Mode 1
- NBS separates, removes, and recycles the entrained moisture and discharges dry steam into the main steam subsystem in Mode 1
- NBS provides the ability to produce the steam flow capacity and conditions to the main turbine for electric power production as defined in the reactor heat balance in Mode 1
- NBS transports steam generated in the RPV to the main turbine for electric power production in Mode 1
- NBS vents the non-condensable gases through the RPV head to the main steam piping in Mode 1
- NBS detects leakage from the RPV O-ring seal in Modes 1, 2, 3, 4, and 5
- Upper reactor internals are removable to facilitate the refueling operation

#### **5.3.3.2 Normal Functions (Safety Category)**

- NBS provides the reactor coolant pressure boundary for the cooling water and steam that prevents the nuclear fuel from overheating in Modes 1, 2, 3, 4, and 5

- NBS provides the core support structure to enable the control rods to stop the nuclear reaction when driven into the core by their respective Hydraulic Control Units (HCU) in Modes 1, 2
- NBS provides the core support structure in Modes 1, 2, 3, 4, 5, and 6
- NBS provides the flow path to enable the core coolant to keep the core cool using natural circulation in Modes 1, 2, 3, 4, 5, and 6 (Note: The control rods provide the negative reactivity into the core to allow for the control of reactor power)
- Main steam has the ability to be isolated independent of offsite power and if the condenser is used for decay heat removal in Mode 4

#### **5.3.3.3 Off-Normal Functions (Non-Safety Category)**

The system does not perform any safety functions during off-normal conditions.

#### **5.3.3.4 Off-Normal Functions (Safety Category)**

The NBS performs DL3 SC1 functions that support high availability of the BWRX-300 Fundamental Safety Functions (FSF) as listed below:

- Control of reactivity
- Fuel Cooling
- Long-term heat removal
- Containment of radioactive materials (isolation of RCPB)

As described in Chapter 3, Subsection 3.1.6, the allocation of defense line functions to system function categorization and system classification is as follows:

- SC1 equipment is located in DL3
- SC2 equipment is located in DL4a
- SC3 equipment is located in DL2

### **5.3.4 System Operations**

#### **5.3.4.1 Initial Configuration (Pre-Startup)**

System configuration is established per plant procedures. Operating modes are defined in Chapter 16, Appendix 16A.

#### **5.3.4.2 System Startup**

Startup of the NBS begins with reactor operations in Mode 2. System startup consists of reactor startup from cold shutdown (RPV coolant bulk temperature less than or equal to 90.0°C) or hot shutdown conditions (bulk temperature greater than 215.6°C). All components and instruments of the NBS are operable and valves are aligned to the positions (open or closed) required.

During startup operation, the NBS is heated to 90°C and condenser vacuum established so the RPV can be deaerated before nuclear heating is started. At this time the MSRIVs and MSCIVs are opened. Reactor startup is begun by withdrawing control rods in a predetermined sequence to achieve core criticality and continued to the onset of coolant boiling. The heat-up rate after boiling onset is automatically controlled using the TBVs and condenser. The reactor water level is controlled by manual operator or automatic direction to maintain the level in the normal range during the transition from SDC to FW injection. Continued control rod withdrawal raises core thermal power up to and not exceeding 12 percent RTP.



RPV level control is performed by regulating the FW flow. With the reactor core power stable at about 12 percent RTP, preparations are made for turbine roll and generator synchronization. The MSL drain line valves and the isolation valves to other system interfaces are open. The MSL drains remain open to approximately 40 percent RTP to remove excess moisture carryover.

#### **5.3.4.3 Normal Operations**

Power Operation (Mode 1) consists of normal operation, planned transients, AOOs and infrequent events (unplanned transients), and on-line testing. Normal operation includes steady-state operation at a given power level up to RTP. All required components and instruments of the NBS are operable and are aligned as required for system operation. Subcooled FW is provided by the CFS system to keep the RPV level in the normal operating range. The reactor pressure is controlled at 7.17 MPa with a constant steam mass flow rate.

As the water boils in the core, the chimney provides an area of expansion such that the velocity lowers and allowing the void fraction to increase. The NBS also provides the means for separating the steam from the liquid and drying the steam by channeling flow through the steam separator and steam dryer before it goes to the MS lines and the turbine. The MS subsystem transports steam generated in the RPV through PCS discharging from the RPV nozzles through the MSRIVs, traveling through the MSLs and MSCIVs to provide the steam to drive the main turbine. Steam is directed through the open maintenance block valves to the MSR, number 6 FWH, and TASS. MS line drain line valves open and close as a function of the level of condensed steam in the drain pot up to about 40 percent power where they are verified closed.

During power operation, the non-condensable gases generated in the RPV will be vented through the RPV head to the MS piping.

#### **5.3.4.4 Off-Normal Operations**

##### ***5.3.4.4.1 Planned Transients***

Planned transients include reduction or increases in power, rod pattern changes, reduction to 0 percent power and hot standby.

Reductions in power to 50 percent by rod pattern changes have no effect on NBS operation.

Reduction in power to 40 percent and below results in automatic opening of the drain orificed line valves to the main condenser.

The TBVs open whenever the TSVs and TCVs close, bypassing steam to the condenser.

##### ***5.3.4.4.2 Unplanned Transients***

Unplanned transients are listed in Chapter 15, Section 15.5.

#### **5.3.4.5 System Shutdown**

System Shutdown contains three (3) states:

- Hot Shutdown
- Stable Shutdown
- Cold Shutdown

##### ***5.3.4.5.1 Hot Shutdown***

During Hot Shutdown (Mode 3), the NBS is intact and available to be pressurized.

Reactor shutdown from the power operation begins with the reactor at rated pressure and temperature and the reactor Mode Switch in RUN position. Rod insertions reduce plant power

until the turbine is taken offline. RPV coolant level is normally controlled using FW to provide reactor water level control if the main condenser is available to reject core decay heat. CUW can be used to overboard excess coolant level if the RPV is pressurized.

Insertion of control rods is performed to establish a reactor cooldown rate and slowly depressurize the reactor. After all control rods are inserted, the reactor is placed in the Hot Shutdown position. If conditions require operator action to expedite core shutdown, the control rods are rapidly inserted by a trip signal initiated by placing the reactor Mode Switch to the SHUTDOWN position.

In Hot Shutdown, the RPV coolant bulk average temperature is less than rated conditions but greater than 215.6°C. The NBS continues to provide steam paths for the removal of core decay heat by the main condenser or by the ICS, depending on the status of the NBS isolation subsystems. Makeup is provided as required either from the CFS or from CRD system.

#### **5.3.4.5.2 Stable Shutdown**

Entering Stable Shutdown (Mode 4), starts with the reactor exiting Hot Shutdown usually before the reactor Mode Switch is placed in the SHUTDOWN position. The NBS is intact and able to be pressurized. Insertion of control rods is continued to establish a reactor cooldown rate and slowly depressurize the reactor. If FW is available, decay heat can be removed using the TBVs to discharge steam to the condenser and return back using FW and/or condensate. MS may or may not be isolated depending on the availability of offsite power and whether the condenser is available for decay heat removal. If offsite power is not available, decay heat removal is performed by the ICS.

#### **5.3.4.5.3 Cold Shutdown**

Normally, the reactor design is to be cooled down from the HOT SHUTDOWN condition by opening one or more TBVs to direct steam to the main condenser. RPV cooldown is carefully controlled so that it is less than the design limit of 55.5°C/hr. SDC initially provides a small assist to this cooling. When the reactor pressure reaches approximately the pressure at which turbine gland sealing steam is no longer effective at maintaining vacuum in the main condenser, the remaining residual heat in the reactor is within the capability of SDC. Both FW lines provide a path for SDC flow to return to the RPV. Reactor cooldown continues with SDC until entering Cold Shutdown (Mode 5) with reactor bulk coolant temperature at or below 93.3°C. Coolant temperature is maintained above the design minimum temperature with the head bolts tensioned. The NBS instrumentation monitors the reactor water level, temperature, and pressure.

### **5.3.5 Refueling Mode Operation**

When the RPV is to be filled for refueling, the RPV is further cooled to near temperature inside containment using SDC. When the reactor vessel head stud de-tensioning begins, the plant operating state is Refueling Mode (Mode 6). The NBS instrumentation is used to monitor the reactor water level and temperature. No further NBS action is needed unless RIV maintenance is planned in which case nozzle plugs are installed.

### **5.3.6 Testing and Maintenance**

This section provides, in part, the basis for preparation of detailed maintenance procedures. It defines the maintenance philosophy, outlines the procedures for scheduled (preventative) and unscheduled (corrective) maintenance, and ISI and surveillance. It also identifies interfacing systems needed to support maintenance operations.

The BWRX-300 in-service inspection requirements specification provides mandatory Pre-Service Inspection, In-Service Inspection (ISI), and In-Service Testing (IST) requirements for components and systems. Pre-Service Inspection, ISI, and IST requirements are consistent

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

with ASME BPVC, Section XI (Reference 5.3-6) and ASME OM, "Operation and Maintenance of Nuclear Power Plants," (Reference 5.3-7).

The Pre-Service Inspection, ISI, and IST requirements include examinations, inspections, and testing of the Reactor Pressure Vessel (RPV) and reactor coolant transporting systems, components, piping, and pipe supports, which are designed and installed in accordance with ASME BPVC, Section III, and applicable codes and standards listed in Chapter 1, Appendix B. In addition, this specification includes basic design control considerations within ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications", Part 2 (Reference 5.3-8). The ISI requirements ensure plant nuclear safety, minimal radiation exposure to the general public, and plant personnel, and proactive management of potential adverse plant system deterioration and aging effects.

Any required surveillance testing is determined as part of the work planning / work release approval process. Post-Maintenance Testing is determined as part of the maintenance review and approval process.

Pre-service and in-service inspection are performed in accordance with ASME BPVC Section XI.

The RIVs have appropriate access platforms and load handling equipment to assist valve assembly maintenance and load transport inside the containment.

The NBS water quality meets the requirements delineated in the BWRX-300 Water Quality Design Specification.

Areas requiring ISI inspection are provided with access spaces/routes and with removable and reusable insulation coverings.

### **5.3.1 References**

- 5.3-1 USNRC NUREG-0800 BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."
- 5.3-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 5.3-3 USNRC NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants."
- 5.3-4 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components- Division 1- Subsection NCD- Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 5.3-5 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.
- 5.3-6 ASME BPVC-XI-1," BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers.
- 5.3-7 ASME OM, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers.
- 5.3-8 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.

## **5.4 Reactor Pressure Vessel**

The BWRX-300 RPV (Figure 5.4-1) assembly consists of the vessel with nozzle integral RIVs and its other appurtenances, a removable head, the reactor internals and supports and instrumentation. The RPV instrumentation that monitors the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the FMCRDs.

The RPV is a vertical, cylindrical pressure vessel fabricated from forged rings or rolled and welded plate rings joined together, with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles including RIVs, and reactor internals support. The reactor vessel is relatively tall which permits natural circulation driving forces to produce core coolant flow.

The reactor pressure vessel is designed to keep its structural integrity during all specified loading conditions. It also maintains its integrity against nonductile failure including consideration of material embrittlement due to neutron irradiation during its design life.

A longer internal recirculation flow path (relative to forced-circulation BWRs) is provided by a “chimney” in the space that extends from the top of the core to the entrance of the steam separator assembly. The top guide, the chimney, chimney head and steam separator assembly are supported by a shroud assembly that extends to the bottom of the core.

### **5.4.1 Major Reactor Internal Components**

This section addresses the RPV internals which consist of all the structural and mechanical elements inside the RPV.

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and in-core nuclear instrumentation) are as follows:

- Core support structures (shroud support, core plate, shroud, top guide, chimney, control rod guide tubes, control rod drive housing, orificed fuel supports and peripheral fuel supports)
- Internal structures (chimney head and steam separator assembly, steam dryer assembly, feedwater spargers, internal piping, nuclear instrumentation, in-core guide tube stabilizers, surveillance sample holders)

A simplified representation of the BWRX-300 RPV and internal component locations is shown in Figure 5.4-1.

### **5.4.2 Core Components (Control Rods and Nuclear Instrumentation)**

Major Core Components including, Fuel Bundles and Control Rods are discussed in Chapter 4. The control rods (see Chapter 4, Figure 4.1-1) insert negative reactivity into the core to control reactor power.

The nuclear (in-core) instrumentation consists of the PRNM, GTs, and WRNM.

The PRNM provides neutron monitoring power signals to the C10 Protection System. The PRNM also provides signals for post-accident monitoring purposes and, through isolated, one-way optical data links to the core monitoring three-dimensional power distribution program and to the DL2 control rod blocking systems.

GTs are an in-core device that converts local gamma flux to an electrical signal; gamma flux and neutron flux are both representative of core thermal power. It represents a completely diverse

technology to the neutron detecting LPRMs used in DL3 PRNM system. They supply information needed to allow for calibration of the LPRMs in the PRNM system.

The WRNM system is a redundant pair of industrial microcomputers with a real time operating system that each monitor all of the WRNM neutron monitoring detectors in the core. The detectors are distributed radially in the core at fixed heights with at least three detectors in each quadrant. The detectors can be bypassed but the bypass switches are arranged such that only one detector can be bypassed at a time in any quadrant. Each detector is sensitive to neutrons from below criticality to approximately 20 percent RTP – a range of over 11 decades. Their absolute value is indicative of reactor power.

### **5.4.3 Core Support Structures**

The basic core support structure components consist of the core support legs, lateral braces, core support ring, core plate, shroud, top guide, chimney, and the orificed fuel supports, peripheral fuel supports, the control rod guide tubes, and non-pressure boundary portion of control rod drive housings.

The core support structure includes the chimney and two basic structures. The chimney is an extended height cylindrical component mounted on the top guide and attached to the top of the shroud. It provides the necessary additional volumetric space within the RPV that permits natural circulation driving forces to produce core coolant flow. The first structure horizontally restrains the entire core and vertically supports the peripheral fuel assemblies. The second structure provides vertical support for the remainder of the core by supporting the fuel assemblies in independent groups of four.

The first structure consists of the support legs, support ring, lateral braces, core plate (with peripheral fuel supports), shroud, top guide, and the hardware necessary to assemble these components. The support legs are affixed to the bottom of the RPV, and the support ring is attached on top of the legs. The support legs and support ring provide a mounting point for the core plate and vertically support the core plate and components above it. Lateral braces are mounted to stiffen the support structure and provide lateral restraint of horizontal loads from the internals and the core.

The second structure consists of the orificed fuel supports, the control rod guide tubes, and the upper portion of the control rod drive housings inside the RPV.

Core support structures perform a safety function and are classified as SC1. Core support structures are Seismic Category A.

The reactor internals are classified as core support structures according to the ASME BPVC, Section III, Division 1, Paragraph NG-1121, Core Support Structures (Reference 5.4-1). They are designed, fabricated, and examined in accordance with the provisions of ASME BPVC Section III, Division 1, Subsection NG (Reference 5.4-1).

The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals (other than the core support structures) meet the guidelines of the ASME BPVC, Section III, Division 1, Subsection NG-3000 (Reference 5.4-1), are constructed so as to not adversely affect the integrity of the core support structures. If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines are identified, and their use justified in the design. For non-ASME Code core structures, components and supports, design margins presented for allowable stress, deformation, and fatigue are equal to or greater than margins for other plants of similar design with successful operating experience. Specific reactor internals of a specific safety class will be designed, fabricated, and examined in accordance with the

applicable codes and standards, such as ASME BPVC Section III, Division 1 for Light Water Reactors (LWR).

#### **5.4.3.1 Shroud**

The SC1 shroud (Figure 5.4-1) is the cylindrical support of the reactor core. It is supported by the shroud support assembly. The core plate is between the shroud and the shroud support, and these three pieces are bolted together. The top guide is between the shroud and the chimney barrel, and these three pieces are bolted together.

#### **5.4.3.2 Shroud Support**

The SC1 shroud support (Figure 5.4-1) is a circular assembly with vertical legs that are welded to the reactor bottom head. The shroud support provides the vertical and lateral support for the shroud, and other components like the top guide and core plate. It also supports the chimney and chimney head and steam separator assembly, and steam dryer assembly.

#### **5.4.3.3 Core Plate**

The SC1 core plate (Figure 5.4-1) is at the bottom of the reactor core area. It provides the lateral support for the fuel assemblies and control rods through the orificed fuel supports, provides vertical and lateral support for the nuclear instrumentation, and provides vertical and lateral support the startup sources, and for the peripheral fuel supports and fuel bundles.

#### **5.4.3.4 Top Guide**

The SC1 top guide (Figure 5.4-1) is at the top of the reactor core and consists of a circular plate with square openings for fuel. It provides the lateral support for the fuel assemblies, nuclear instrumentation, startup sources, and control rods.

The top guide is mechanically attached to the top of the shroud and provides a flat surface for the chimney flange.

#### **5.4.3.5 Control Rod Guide Tube**

The SC1 control rod guide tubes (Figure 5.4-1) fit in holes in the core plate and rests on the control rod housings that are welded to the reactor bottom head. They provide the lateral support by these upper and lower intersections. The control rod guide tubes channel water to the control rods as they move up and down within the core to change the reactor power levels. The control rod guide tubes also provide vertical support to the orificed fuel supports and to four fuel bundles landed on each orificed fuel supports.

#### **5.4.3.6 Control Rod Drive Housing**

The SC1 control rod drive housings provide an extension of the RPV pressure boundary at the bottom head for installation of FMCRDs (Figure 5.4-1). They also support the weight of a control rods, FMCRDs, control rod guide tubes, orificed fuel supports, and four fuel assemblies per orificed fuel supports. Finally, the control rod drive housing flanges also provide the ports for the attachment of CRD hydraulic supply lines.

#### **5.4.3.7 Orificed and Peripheral Fuel Supports**

The SC1 orificed fuel supports (Figure 5.4-1) rest inside the respective control rod guide tubes with a flanged top end that sets down onto the control rod guide tubes top end that transfers vertical load. They provide a slot for the lower end of each fuel assembly to rest and, also provide cruciform slots to guide the control rod blade as it raises or lowers to respectively increase or decrease. The orificed fuel supports slots are crucial to keeping the control rods properly aligned between the four bundles for rapid insertion into the core from fully withdrawn.

The peripheral fuel supports rest in the core plate around the outside of the orificed fuel supports and fuel bundles. A single fuel bundle rests in each peripheral fuel support. Their function is the same as an orificed fuel supports.

#### **5.4.4 Chimney**

The SC1 chimney (Figure 5.4-1) is a long cylinder mounted to the top guide that supports the steam separator assembly. The chimney forms the annulus separating the subcooled recirculation downward flow from the upward steam-water mixture flow exiting the core. The recirculation flow consists of reactor coolant returning from the steam separators and feedwater makeup. The cylinder is flanged at the bottom and top for attachment to the top guide and the chimney head, respectively. Lateral support brackets are located around the periphery of the chimney exterior beneath the flange and attached at the vessel shell internal surface. These brackets brace the chimney and the chimney head – steam separator assembly against horizontal motion.

The chimney is a unique feature to natural-circulation BWRs. It is used to extend the area of the RPV above the core and below the separator to increase the thermal driving head for natural circulation (and thereby drive a higher core flow). The OPEX with chimneys in natural-circulation BWRs is from several reactors such as the Vallecitos BWR, Humboldt Bay, and Dodewaard. The previous natural-circulation BWRs designs used chimneys with flow partitions. For the BWRX-300 chimney design, the chimney partitions are removed, which simplifies construction and installation, eliminates the need to remove the chimney during refueling outages, simplifies core alterations, and shortens the overall outage schedule.

#### **5.4.5 Chimney Head and Steam Separator Assembly**

The chimney head and steam separator assembly does not perform a safety function and provides a path for the steam flowing from the core to the steam dryer assembly. The steam separator part of the assembly uses centrifugal force to decrease most of the moisture in the steam before it enters the steam dryer assembly.

#### **5.4.6 Steam Dryer Assembly**

The steam dryer assembly does not perform a safety function and completes the task of removing the moisture from the steam before it exits the reactor. The moisture content is lower than 0.1 percent at 100 percent RTP when the steam leaves the reactor on its way to the main high-pressure turbine.

The steam dryer is supported by a set of brackets equally spaced around the vessel shell internal wall. The dryer support ring rests on these brackets. Additional hold-down brackets are mounted inside the vessel head to restrain vertical movement of the dryer under seismic or dynamic loads. These hold-down brackets are installed so that they are over the lifting lugs of the steam dryer. Horizontal motion due to seismic or dynamic loads is limited by seismic blocks mounted to the steam dryer support ring.

#### **5.4.7 Feedwater Spargers**

The feedwater spargers do not perform a safety function. Each of two feedwater lines is connected to spargers through four RPV nozzles. The feedwater spargers deliver makeup water to the reactor during plant start up, power generation and plant shutdown modes of operation.

#### **5.4.8 Internal Water Piping**

The reactor contains two internal suction lines that are routed down to the bottom head area of the reactor from the two N5 reactor nozzles. These two nozzles are located above TAF+4m, which is consistent with the BWRX-300 design requirement for minimum nozzle height.

The two internal suction lines are routed from flange connection inside the vessel for each respective nozzle, through an inverted loop, and down the vessel shell internal wall to the periphery of the CRD penetrations in the RPV bottom head. The piping for each internal drain line is attached to the vessel shell internal wall by brackets, spaced to provide support for anticipated loads and to inhibit pipe fatigue due to flow induced vibration. The brackets are designed to the rules ASME BPVC Section III, Division 1, Subsection NF (Reference 5.4-2), categorized as SC1, and Seismic Category A.

Each of the two suction lines also contain a vacuum breaker hole at the top of the pipe loop immediately inside of the two nozzles to prevent inadvertent draining. These two suction lines are connected externally to the reactor through the associated RIVs to the CUW. The two suction lines internal to the reactor allow the CUW to remove water from the RPV bottom head area.

#### **5.4.9 Reactor Pressure Vessel Head Vent**

The SC1 internal vent piping ducts non-condensable gas flow down the inside of the RPV closure head, into a discharge line that is integrated into and through the bracket of one of the internals guide rods, and to the discharge through the head vent nozzle. The internal piping is, thus, split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The piping in the RPV closure head is mounted with brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The head vent piping penetrates the RPV through a nozzle on the side of the reactor with two RIVs flange-mounted in series. These RIVs are SC1 and perform the RPV single failure proof automatic isolation function.

#### **5.4.10 Nuclear Instrumentation Tubes and Stabilizers**

The nuclear instrumentation tubes are SC1 components, and the stabilizers do not perform a safety function. The tubes protect the nuclear instrumentation from the flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core. The nuclear instrumentation tubes extend from the bottom head to the top guide. The PRNM detectors and the WRNM detectors are inserted through the guide tubes.

A latticework below the core plate consisting of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes.

#### **5.4.11 Surveillance Sample Holders**

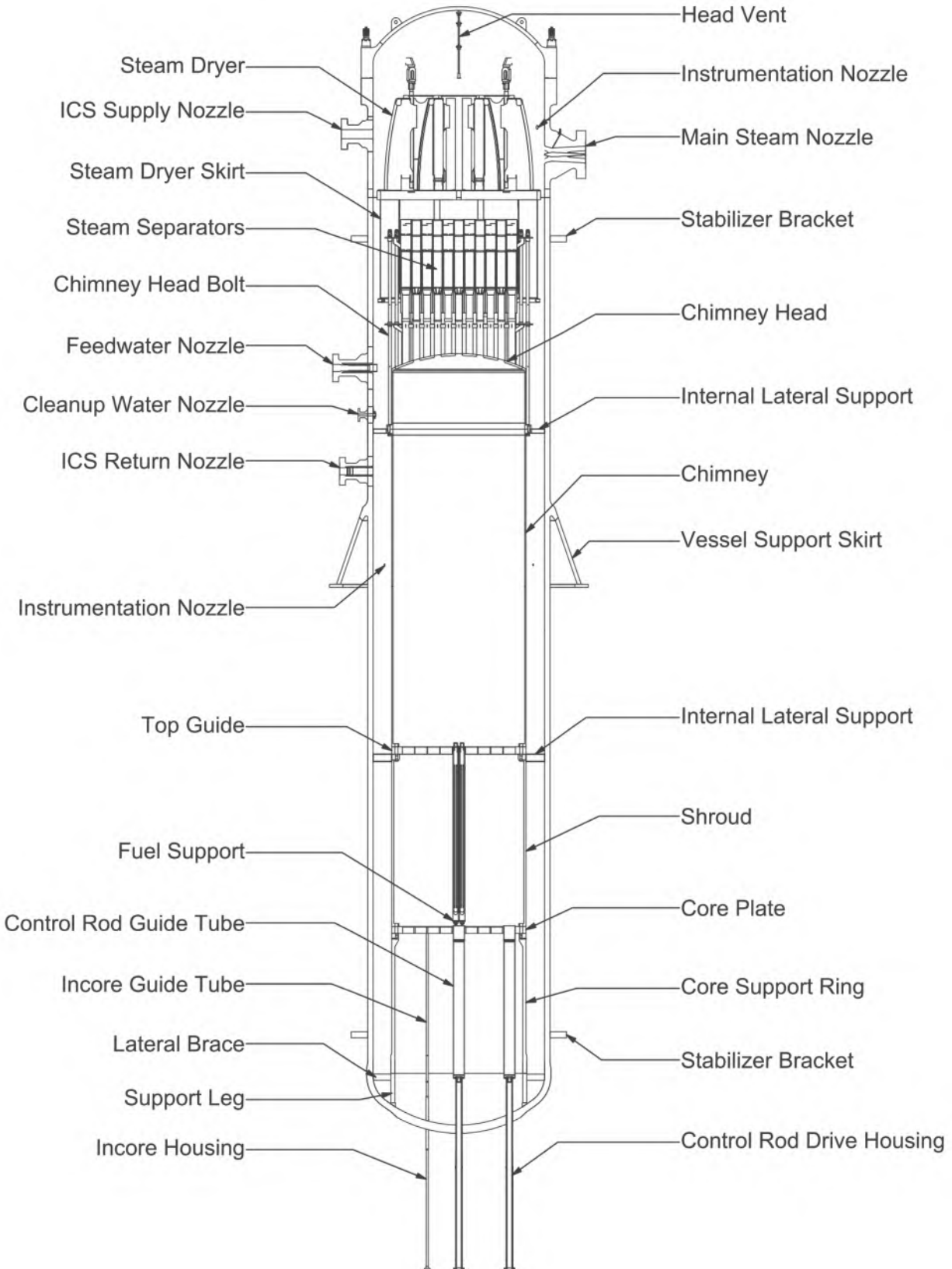
The surveillance sample holders are welded baskets containing impact, notch-sensitivity, and tensile specimen capsules. The baskets hang from brackets that are attached to the inside of the reactor vessel wall and centered mid-height of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron fluence experienced by the reactor vessel itself.

#### **5.4.12 References**

- 5.4-1 ASME BPVC, "Section III Division 1 Subsection NG, Rules for Construction of Nuclear Facility Components Core Support Structures," American Society of Mechanical Engineers.
- 5.4-2 ASME BPVC, "Section III, Division 1, Subsection NF, Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers



NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 5.4-1: BWRX-300 Reactor Pressure Vessel and Internals**

## **5.5 Reactor Coolant Pumps or Recirculation Pumps**

Reactor coolant flow through the BWRX-300 core is by natural circulation, meaning pumps (reactor coolant or recirculation) are not required to force reactor coolant through the core. Natural circulation is enabled mainly by the addition of a high chimney between the top of the core at the top plate to the bottom of the steam separator.

## **5.6 Reactor Coolant Piping**

There is no specific ASME BPVC Section III, Division 1, Class 1 reactor coolant piping as part of the NBS. The BWRX-300 RCPB is described in Subsection 5.1.1. Following insertion of control rods, the BWRX-300 cools and depressurizes the reactor by natural circulation with decay heat removal by the ICS or balance of plant. During overpressure events or loss of normal heat sink events, the ICS is used to quickly depressurize the RCPB and cool the core.

The ICS is designed to both ASME BPVC Section III, Division 1, Subsection NB (Class 1) (Reference 5.6-1) and ASME BPVC Section III, Division 1, Subsection NCD (Class 2) (Reference 5.6-2) as described in Chapter 6, Subsection 6.2.3.

### **5.6.1 References**

- 5.6-1 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 5.6-2 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.

## **5.7 Reactor Pressure Control**

### **5.7.1 System and Equipment Functions**

This section describes systems that protect the RCPB from overpressurization. Defense Lines (DL) 1 through 4 are discussed in Chapter 3, Section 3.1.

DL1 operational state is normal operation (prevention of abnormal operation and failures) where pressure control is provided by modulating the Turbine Control Valves (TCVs).

DL2 operational state occurs during AOOs (control of abnormal operation and detection of failures) for which the plant design is to limit pressure deviations and restore normal pressure.

SC3 I&C control system regulates RPV pressure by controlling the position of the TCVs or Turbine Bypass Valves (TBVs).

During plant startup, once boiling commences, the RPV continues to heat up and pressurize to the rated pressure. The TBVs are initially used to control pressure with the NBS kept at saturated conditions. As power increases, the TCVs are gradually opened causing the TBVs to gradually close.

The RPV pressure is normally held constant by positioning the TCVs, so the turbine steam flow always follows the reactor. The TCVs are modulated by their control system to regulate the demand steam flow.

Further discussion on pressure control functions is provided in Chapter 7, Section 7.3.

### **5.7.2 Safety Design Bases**

Overpressure protection for the RCPB is provided through a combination of the NBS, the ICS, and SC1 I&C control system to limit peak pressure during AOOs to less than or equal to 110 percent of RPV design pressure as defined by the ASME BPVC, Section III, Division 1, Article NB-7000, subparagraphs NB-7120(b) and NB-7120(c) (Reference 5.7-1).

Refer to Chapter 6, Section 6.2 for overpressure engineered safeguard functions.

### **5.7.3 Description**

Reactor pressure is controlled from a cold vessel through pressurization and heat up to power operation by the SC3 I&C control system for reactor pressure control using the TBVs and TCVs. The SC3 I&C control system during normal power operation controls reactor pressure using the TCVs. Refer to Chapter 10, Subsections 10.2.6 and 10.7.6 for normal operation and startup, respectively.

The SC3 I&C System Isolation Condenser (IC) Throttling Pressure Controller performs the functions to prevent exceeding the reactor cooldown rate. Detailed description is provided in Chapter 7, Section 7.3.

### **5.7.4 Interfaces with Other Equipment or Systems**

Figure 5.1-1 NBS Input-Output provides the NBS interfaces with other systems and equipment.

### **5.7.5 System and Equipment Operation**

Chapter 7, Subsection 7.3.3 describes normal pressure control provided by the TCVs. Chapter 6, Subsection 6.2.2 describes the overpressure engineered safeguard features and functions.

### **5.7.6 Instrumentation and Control**

The SC1, SC2 and SC3 I&C Systems (refer to Chapter 7, Section 7.3) provide control of RPV pressure using NBS pressure sensing instrumentation as described in Subsection 5.3.1.3.

#### **5.7.6.1 Leak Detection**

Means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. The components of the RCPB, including the ICS and reactor isolation valves are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with recognized codes and standards, and under an approved quality assurance program with approved control of records.

#### **5.7.6.2 Reactor Pressure Vessel Flange Leak Detection**

The RPV head/shell flange detects leakage any time the reactor is in Operating Modes Power Operation, Startup, and Hot Shutdown.

#### **5.7.7 Monitoring, Inspection, Testing, and Maintenance**

Section 5.11 describes the pre-service and in-service inspection and system pressure test programs for NRC Quality Group A, ASME BPVC, Division 1, Class 1 piping, and components. It describes these programs implementing the requirements of Subsection IWB of the ASME BPVC Section XI Division 1 (Reference 5.7-2).

#### **5.7.8 Radiological Aspects**

Chapter 12, Subsection 12.3.8.2 describes the various system and major component design considerations of the NBS related to plant radiation protection. Chapter 12, Subsection 12.1.5.4 provides information the design provisions and measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA) in operational states and in accident or post-accident conditions.

The radiological consequences of large ICS breaks outside containment are discussed in Chapter 15, Subsection 15.5.9.2.3 while small ICS line breaks are described in Chapter 15, Subsection 15.5.9.2.4. The events considered do not result in any fuel failures or any release of primary coolant to the environment; therefore, there are no radiological consequence associated with these events.

#### **5.7.9 Performance and Safety Evaluation**

This section evaluates systems that protect the RCPB from over pressurization. DL2 is an AOO state for which the plant is designed to limit pressure deviations and restore normal pressure. The evaluation of overpressure control in this section satisfies the AOO response. The bounding events for AOO pressure increase mitigation are described in Chapter 15, Subsection 15.5.3.

The BWRX-300 DSA Deterministic Safety Analysis, presented in Chapter 15, Section 15.5, is the analysis of record for responses to off-normal events and provides the information required to demonstrate that regulatory acceptance criteria are met. The selected bounding events resulting in the most significant challenges to the fission product barriers (including the RCPB) are summarized in Table 15.2-3. LOCA postulated initiating events where piping breaks occur in high-pressure pipes connected to the RPV, are discussed in Subsection 15.5.4.6.

The use of an ICS design for BWRX-300 overpressure protection addresses the need for a reliable pressure mitigation which enhances the safety of the design. This is demonstrated by the safety analysis in Chapter 15, Subsection 15.5.4 which concludes that there is no radiological release consequence for postulated pressure increase events.

The reactor integrity is assured by meeting the pressure criteria, provided in Chapter 15, Section 15.3. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal

water reaction. The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel or pressure vessel. Therefore, these barriers maintain their integrity and function as designed.

#### **5.7.10 References**

- 5.7-1 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 5.7-2 ASME BPVC-XI-1," BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers.

## **5.8 Isolation Condenser System**

The ICS is part of the RCPB and provides a DL barrier for overpressure due to AOO, DBA, and DEC events. If an MSL break occurs or the normal reactor cooling path is unavailable during an AOO or accident response, the ICS is credited for cooling and pressure protection of the RPV and the RCPB. Chapter 6, Subsection 6.2.1 describes the engineered safety function and operation provided of the ICS in mitigating overpressurization of the RCPB.

## **5.9 Reactor Pressure Vessel Component Supports and Restraints**

### **5.9.1 Reactor Pressure Vessel Pedestal**

The RPV pedestal is equipped with a RPV Bracket, where the RPV skirt is anchored using anchor bolts. The RPV pedestal provides structural support for the lower RPV Stabilizers and for the upper containment access platform module which supports the upper RPV Stabilizers. The RPV pedestal is a cylindrical, steel-plate composite structure that supports the RPV. The RPV pedestal, enclosed within the primary containment, is welded to a steel-plate composite basemat that is connected with the Steel-Plate Composite Containment Vessel (SCCV). For additional description of the RPV pedestal refer to Chapter 9B.2.2.

### **5.9.2 Reactor Pressure Vessel Pedestal Safety Basis**

The primary safety functions of the RPV pedestal are as follows:

- The RPV pedestal provides structural support to SSCs such as the RPV, RPV Stabilizers and miscellaneous platforms.
- The RPV pedestal provides some radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECAs.

The RPV pedestal is a SC1 and Seismic Category A structure.

The RPV pedestal does not serve any pressure-retaining function and is outside the scope of ASME Code applicability. Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 containment internal structures are listed in Chapter 1, Appendix B.

Chapter 3, Subsection 3.6.3 discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of the ASME Section III, Division 1.

### **5.9.3 Reactor Pressure Vessel External Supports**

#### **5.9.3.1 Reactor Pressure Vessel Support Skirt**

The RPV is supported above the vessel core beltline region and has stabilizer brackets near the top and the bottom of the vessel that provide lateral support. The support system is a support skirt or apron-type that is connected to an attachment on the cylindrical portion of the vessel. The support skirt attachment knuckle is integrally forged with the vessel cylinder. The PCS provides vertical and lateral RPV support. The PCS is completely enclosed within the deeply embedded RB, and also includes containment penetrations and other safety components. The PCS houses the steel-plate composite internal pedestal that supports the RPV, and also acts as a biological shield.

The vessel support skirt is a truncated conical support appurtenance of the RPV. The support skirt is designed to the rules ASME BPVC Section III, Division 1, Subsection NF (Reference 5.9-1), and is categorized as SC1 and Seismic Category A. The support skirt is attached to the cylindrical shell of the vessel by a weld joint to a circumferential knuckle forged into the vessel. The bottom of the skirt has an attached ring that forms a foot which rests on a structural support plate and transmits the vertical vessel load to the vessel support pedestal. The connection is designed to allow for radial movement due to thermal expansion and contraction of the vessel shell during operational heat-up and cooldown evolutions.



#### **5.9.3.2 Reactor Pressure Vessel Stabilizers**

The BWRX-300 has two sets of horizontal motion stabilizers with a set of stabilizers located above the RPV bottom head and a second set of six stabilizers located below the main steam nozzles. The stabilizer design is based on existing proven hardware design from the BWRs operating experience. The stabilizers are designed to the rules ASME BPVC Section III, Division 1, Subsection NF, and are categorized as SC1 and Seismic Category A. The main function of the stabilizers is to provide horizontal support to limit vessel motion by transferring load from the vessel to the adjacent civil structure. The upper and lower vessel stabilizers function independently, depending on the direction of vessel motion due to an applied dynamic force, but horizontally and vertically parallel stabilizers will act with combined resistance against vessel motion.

Refer to Chapter 9B, Subsection 9B.2.2 for further details.

#### **5.9.4 Reactor Pressure Vessel Internal Supports**

##### **5.9.4.1 Chimney Support Brackets**

Lateral support brackets are located around the periphery of the chimney barrel exterior beneath the flange and attached at the vessel shell internal surface. These brackets brace the chimney and the chimney head – steam separator assembly against horizontal motion. The brackets are designed to the rules ASME BPVC Section I, Division 1, Subsection NF, categorized as SC1, and Seismic Category A.

##### **5.9.4.2 Steam Dryer Support Brackets**

The steam dryer is supported by a set of brackets equally spaced around the vessel shell internal wall. The dryer support ring rests on these brackets. Additional hold-down brackets are mounted inside the vessel head to restrain vertical movement of the dryer under seismic or dynamic loads. These hold-down brackets are installed so that they are over the lifting lugs of the steam dryer. Horizontal motion due to seismic or dynamic loads is limited by a set of seismic blocks mounted to the steam dryer support ring. The brackets are designed to the rules ASME BPVC Section III, Division 1, Subsection NF, categorized as Safety Class SC1, and Seismic Category A.

##### **5.9.4.3 Internal Piping Support Brackets**

###### ***5.9.4.3.1 Reactor Water Cleanup System Internal Piping***

Two CUW internal suction lines are routed from flange connection inside the vessel to each respective nozzle, through inverted loops, down the vessel shell internal wall, and to the RPV bottom head. The piping for each internal suction line is attached to the vessel shell internal wall by brackets, spaced to provide support for anticipated loads and to inhibit pipe fatigue due to flow induced vibration. The brackets are designed to the rules ASME BPVC Section III, Division 1, Subsection NF, categorized as SC1, and Seismic Category A.

###### ***5.9.4.3.2 Head Vent Internal Piping***

The internal head vent piping inside the RPV closure head is a discharge line that is integrated into and through the bracket of one of the guide rods. The internal piping is split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The piping in the RPV closure head is mounted with brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The brackets are designed to the rules ASME BPVC Section III, Division 1, Subsection NF, categorized as SC1, and Seismic Category A.

### **5.9.5 Materials**

Refer to Table 5.2-3 for reactor internal components materials listing.

### **5.9.6 Interfaces with Other Equipment or Systems**

The BWRX-300 RPV pedestal and bioshield are integrated with the RB and PCS common mat foundation as shown in Figure 9B-1 and the RPV pedestal interfaces with the RPV through the RPV skirt support and horizontal stabilizers. Further description is provided in Chapter 9B, Subsection 9B.2.2.5.

### **5.9.7 Monitoring, Inspection, Testing, and Maintenance**

RPV internal and external component supports are subject to an inspection program as described in Section 5.11.

As described in Chapter 3, Subsection 3.5.4.6, a formal program for monitoring, testing and inspection is not required for PCS internal civil structures.

### **5.9.8 Radiological Aspects**

Refer to Subsection 5.7.8.

### **5.9.9 Performance and Safety Evaluation**

The RPV support design uses a conical skirt to support vertical load and horizontal stabilizers to minimize vessel motion due to horizontal load. The design is based on long operating experience with supports of the same basic design. The RPV support skirt design concept is a standard textbook method for mounting a pressure vessel that is employed in industries other than commercial nuclear power. The conical skirt shape permits the mid-vessel mounting of the skirt with the support ring foot set on the bracket attached to the cylindrical RPV pedestal. The support skirt is the primary interface with the civil structure supporting the RPV assembly and a transfer point for seismic accelerations from the RB to the RPV assembly. The support skirt and horizontal stabilizer brackets are SA-508M, Grade 3, Class 1 material to match the RPV material. The support skirt flange is integral to the RPV shell forging. The horizontal stabilizer assembly is predominantly fabricated from SA-105M carbon steel forgings.

The load of the BWRX-300 is more complex because of the relatively high locations of the vessel nozzles relative to the core and that there are large valves directly mounted to the vessel shell. Generally, BWR vessel load is concentrated below the vertical axis center point because of the mass of the core and the large liquid water volume in the lower vessel regions. However, the BWRX-300 has an additional large load concentration above the centerline because of the RIVs. The combined low and high mass centers result in a relatively higher RPV assembly mass centroid in the BWRX-300 design.

It is also known that RPV assembly response to dynamic or seismic accelerations in the horizontal direction is lowered by placing the attachment of the support skirt closer to the centroid of the RPV assembly mass (Reference 5.9-2). The BWRX-300 RPV support skirt is attached at a mid-vessel elevation to place it much closer to the centroid than would be achieved with a traditional skirt attachment that is just above the bottom head. Because of the height of the natural-circulation RPV (which is taller than a forced-circulation RPV due to the additional chimney component height) the BWRX-300 RPV is equipped with both upper and lower sets of horizontal stabilizers. The dual stabilizer sets design provides more control of vessel horizontal motion to limit vessel shell twisting or rocking around the skirt attachment. This limitation on motion also mitigates the dynamic input loads to the RPV nozzles to control the stress on the nozzle connections to the vessel shell. Further, limiting input motion to the RPV nozzles also limits the

acceleration input into the RIV masses and provides greater assurance of reliable RIV function during postulated seismic and dynamic events.

The internal components support brackets are sized based on the static loads of the supported components. Mechanical analysis is performed using loads and loading combinations based on those stated in Chapter 3 Table 3.6-2. The analysis results are used to confirm the bracket sizing of the internal components.

#### **5.9.10 References**

- 5.9-1 5.9-1 ASME BPVC, "Section III, Division 1, Subsection NF, Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers.
- 5.9-2 Henry H. Bednar, "Pressure Vessel Design Handbook," Krieger Publishing Company, Malabar, FL, 1986.

## **5.10 Nuclear Boiler System Connected Valves**

The BWRX-300 design focuses on the mitigation of LOCAs by reducing the number and size of RPV nozzles as compared to previous designs and limiting nozzle penetrations to the upper regions of the vessel. The RPV has integral isolation valves attached directly to the RPV as appurtenances for piping systems greater than DN20 nominal pipe diameter. The BWRX-300 design also includes an outboard MSCIV on each MSL.

### **5.10.1 Reactor Isolation Valves**

The NBS RIVs are part of a fail-safe system designed to isolate the RPV during Service Levels A through D under the full range of reactor pressures and flows. The isolation system consists of the MSRIVs, FWRIVs, ICRRIVs, ICSRIVs, CUW RIVs and the RPV Head Vent RIVs. All of the RIVs are classified as SC1.

RIVs and CIVs function to limit the loss-of-coolant from large and medium pipe breaks. Small pipe breaks are defined as those lines size DN20 and under.

Each NBS RIV assembly is connected to the outboard piping using bolted flange connections. The RIVs are connected directly to the reactor vessel using bolted flange connections and are classified as break exclusion areas. All of the RIVs except for those to / from ICS are fail-closed type valves.

The design for these valves is two valves in series that are independently able to isolate the respective nozzle. The RIVs are designed in accordance with the rules and requirements of ASME BPVC, Section III, Division 1, Subsection NB, Class 1 Components (Reference 5.10-1) and Seismic Category B.

This design feature allows the reactor to be isolated, quickly stopping leakage from the vessel following a transient or downstream pipe break. The BWRX-300 Reactor Isolation Valve design considers the following design features:

- Isolation Valve integral to each RPV large-bore process nozzle
- Redundant, leak proof valves
- Passively or automatically activated

Figure 5.10-1 shows an example of RIV double valve bodies attached to the RPV using flange connections

### **5.10.2 Main Steam Containment Isolation Valves**

The BWRX-300 design includes an outboard MSCIV on each MSL. The MSCIVs provide the SC1 single failure proof isolation of the PCS in the event of accidents or other conditions and prevent the unfiltered release of containment contents. This containment isolation is single failure proof. The MSCIVs are fast-closing and fail-closed type valves. The MSCIVs are fast-closing and fail-closed type valves.

These isolation valves outside of containment are located as close to the containment as practical to satisfy the PCS isolation requirements.

The MSCIVs are designed in accordance with the rules and requirements of ASME BPVC, Section III, Division 1, Subsection NCD, Class 2 and 3 Components (Reference 5.10-2), in accordance with their quality group classification. The MSCIVs are designed for Seismic Category B and certified by a qualification program prepared and performed in accordance with ASME Standard QME-1 (Reference 5.10-3).

The MSRIVs and MSCIVs provide isolation of the MSLs. MSCIVs provide containment isolation, when required during plant shutdown conditions.

All CIVs have individual leakage testing performed to support the overall requirement to verify containment leakage is within the allowable limits. Chapter 6, Subsection 6.3.4 describes containment isolation in detail

### **5.10.3 Safety Design Bases**

The RPV isolation function is designed as SC1 in accordance with the plant design requirement. All of the RIVs except those for the ICS nozzles have fail-closed type actuators with automatic isolation signals. The ICS nozzles must remain open during events requiring ICS to function and use fail-as-is actuators but have individual train automatic isolation on detection of a pipe leak or break in the train. The RIVs of each RPV nozzle are installed using flanged connections and mounted in a double valve body or two valves in a tandem configuration.

The RIVs also constitute the inboard CIV, where applicable, to provide the double isolation of the piping PCS penetration.

The ICS RIVs are normally open during operation. Isolation during on-line operation is normally controlled by the ICS condensate return valves. If called upon by an isolation signal, all other RIVs close. The ICS RIVs are designed to fail as-is, which is the normally open position. Flow will be initiated by the opening of the ICS condensate return valves. The only time the fail as-is position would be non-conservative is if the ICS RIVs are closed for maintenance but that would already be accounted for by the planning of the maintenance in accordance with the Operating Limits and Conditions (OLC).

The BWRX-300 design provides direct position indication of the RIVs. The ICS response time of each required division will be verified within design limits per the IST program. Figure 5.10-1 shows an example of RIV double valve bodies attached to the RPV using flange connections.

Design of the CIVs, with the exception of the RIVs, meets the rules and requirements of ASME BPVC, Section III, Division 1, Subsection NCD, Class 2 and 3 Components.

### **5.10.4 Materials**

Materials are provided in Section 5.2.1. Material specifications used for the RIVs and the CIVs are in accordance with approved processes and with the rules and requirements of ASME BPVC, Section II, Material Specifications (Reference 5.10-4).

### **5.10.5 Interfaces with Other Equipment or Systems**

Figure 5.1-2 provides the interfaces with the RPV.

### **5.10.6 Instrumentation and Control**

Each NBS MSRIV and MSCIV have automatic and manual control capability, with separate pilot valves, and independent test logic control.

The SC1 close function of the MSRIVs and MSCIVs is independent from the other control architecture.

Each NBS MSRIV and MSCIV is capable of isolating the main steam piping in the event of a LOCA (or other events requiring containment or system isolation) to limit the release of reactor coolant.

#### **5.10.7 Monitoring, Inspection, Testing, and Maintenance**

NBS power-operated valves inside the RB are equipped for remote manual functional testing from the Main Control Room (MCR). The ASME Class 1 valves undergo functional testing using equipment qualification methods that examine functionality and structural capacity under seismic and AOO loads, and accident loading conditions. Functional testing evaluates any identified leakage found during post-test inspections.

NBS is designed with provisions for initial and periodic testing of the ASME Code Class 1 and Class 2 system equipment, including hydrostatic test in accordance with the requirements of ASME BPVC Section III, Division 1.

#### **5.10.8 Radiological Aspects**

Refer to Subsection 5.7.8.

#### **5.10.9 Performance and Safety Evaluation**

All NBS RIVs fail-closed except for the ICS RIVs which fail as-is. The ICS RIVs fail as-is because they perform safety functions during DBAs. The ICS RIVs automatic closure function relies on the SC1 I&C system when initiated by the leak detection instrumentation. The NBS CIVs fail-closed design provides conservative isolation of the PCS.

For small pipe breaks, the BWRX-300 design provides a substantial reservoir of water above the core that is sufficient to ensure RPV water level is maintained at or above a safe level following a LOCA. For larger pipe breaks, the respective RIVs, dependent on the location of the pipe break, will close rapidly to prevent significant loss of RPV inventory. In conjunction with RPV isolation, ICS will be actuated to maintain adequate core cooling. If the large break is located on an ICS line, the RIV associated with the train containing the line break will close. If the large break occurs inside containment, the MS CIVs and MSL RIVs will close to prevent unacceptable releases of containment contents.

The RPV RIVs and MSCIVs are single failure proof. The actuation signals for the RIVs are diverse. The RIV design includes redundancy in I&C components and features resulting in a worst-case single failure impacting only one RIV at each RPV nozzle. Diversity for penetrations where RIVs are credited as one of the CIVs have separate and diverse control systems that are single failure proof.

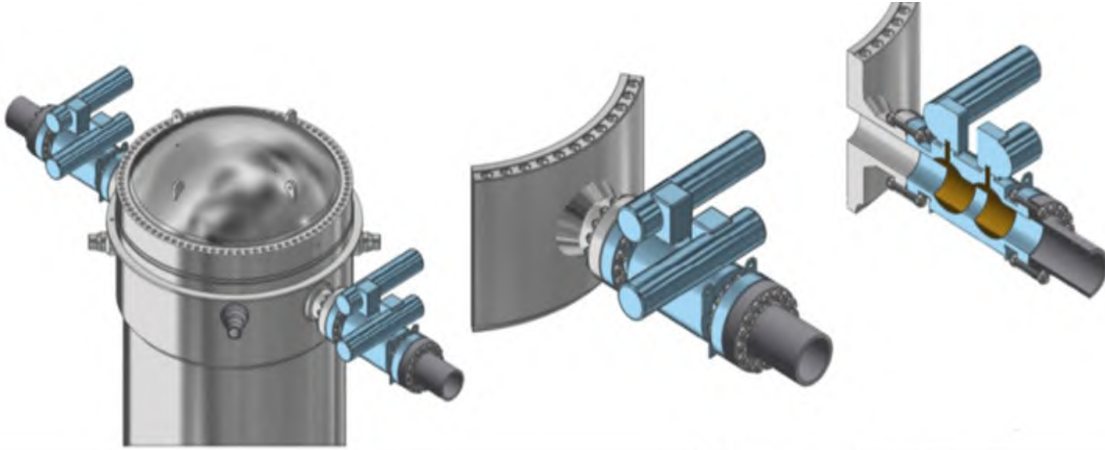
Chapter 15, Subsection 15.5.4 evaluates the bounding DBAs. The scenarios for LOCA events are described in Chapter 15, Subsection 15.5.4.6 and conclude that during large break LOCAs, RPV inventory loss does not threaten fuel integrity. After RPV isolation, decay heat is removed by the ICS from the RPV. The LOCA analyses demonstrate that either the core remains covered, or fuel cladding temperature remains below the normal operating temperature for at least 72 hours using conservative assumptions for unisolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation. The evaluation results for non-LOCA events demonstrate significant margin to the acceptance criteria, and barriers maintain integrity and functionality.

#### **5.10.10 References**

- 5.10-1 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 5.10-2 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

- 5.10-3 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.
- 5.10-4 ASME BPVC-XI-1," BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers.



**Figure 5.10-1: Reactor Pressure Vessel Isolation Valve Assembly (Example)**



## **5.11 Access and Equipment Requirements for Inspection and Maintenance**

The NBS is designed to provide adequate equipment removal paths and personnel access for maintenance, inspection, and replacement of components. The overall layout of mechanical equipment, piping, valves, and instrumentation is arranged so as not to interfere with operation and maintenance activities.

Pre-Service Inspection and In-Service Inspection (ISI) requirements for components and systems for the BWRX-300 are consistent with ASME BPVC, Section XI, Division 1 (Reference 5.11-1). In-Service Testing (IST) requirements for components and systems for the BWRX-300 are consistent with ASME OM Code (Reference 5.11-2). These Pre-Service Inspection, ISI, and IST requirements include examinations, inspections, and testing of the Reactor Pressure Vessel (RPV) and reactor coolant transporting systems, components, piping, and pipe supports, which are designed and installed in accordance with ASME BPVC, Section III, Division 1 and applicable codes in Chapter 1, Appendix B.

### **5.11.1 Accessibility**

ASME BPVC Section III Class 1, 2, and 3 mechanical components and equipment (e.g., heat exchangers, pipe supports, pumps, valves, and vessels) are designed with accessible openings for ISI testing, which supports evaluations that justify the operational readiness of components and equipment.

Manufacturers of components and equipment, which require inspections and examinations to satisfy ASME BPVC Section XI, Division 1 requirements, are examined by appropriate inspection and testing methods, as applicable per the ASME BPVC Section III and ASME OM Code.

The RPV engineering design includes provisions for the following ISI access requirements:

1. Access to the annulus between the reactor pedestal wall and vessel is provided from below the vessel by provision in the bottom head insulation to access this region.
2. Access to the RPV exterior above the RPV pedestal wall is provided by provision of removable insulation to access inspection regions.
3. Removable insulation is provided on NBS piping and valves, which extends to the MSCIV.

### **5.11.2 Reactor Coolant Pressure Boundary Components**

The components, which form the RCPB, are designed to permit the following:

1. Periodic inspection and testing of important areas (such as pressure boundary welds and flanges during the post-outage leakage test) and features, to assess their leak-tightness and structural integrity
2. Establish a material surveillance program for the RPV

### **5.11.3 Reactor Coolant System and Connected Systems**

The Reactor Coolant System (RCS) is defined as the components necessary to provide and maintain adequate core cooling conditions (pressure, temperature, and coolant flow rate) for the fuel in power operation. It includes the RPV, the MSLs, and feedwater lines up to and including the outermost CIV. These components form a major portion of the RCPB as described in Section 5.10.

Isolation and maintenance devices and valves are provided in the system to allow for maintenance of the RCS components.

NEDO-33954 REVISION 0  
NON-PROPRIETARY INFORMATION

The following list contains the more important maintenance operations that must be performed on the RCS components:

1. Servicing of RIVs and CIVs
2. Servicing and replacement of level, temperature, and pressure sensors
3. Servicing and replacement of solenoid elements in valve and pilot operators
4. Adjustment and servicing of valve operators, including limit and position switches
5. Cleaning drain line strainers
6. Adjustment and replacement of valve stem packing
7. Major overhaul of all power-operated valves and hydraulically operated valves

Sufficient space, and the additional process system component facilities, is provided between penetration isolation valves and the PCS boundary to permit:

In-service inspection of non-isolable welds:

- In-service inspection of non-isolable welds
- Access and facilities to perform local leak rate testing of isolation valves
- Access to operate local manual controls
- Access to perform isolation valve assembly maintenance
- Cutout and replacement of isolation valves using standard pipe cutting equipment, pipe fitting tools and equipment, and piping component welding equipment

ASME BPVC Section XI, Division 1, Class 1, and Class 2 RCS equipment is designed and arranged to provide ISI accessibility.

Areas requiring ISI inspection are provided with access spaces/routes and with removable and reusable insulation coverings.

#### **5.11.4 Examination Categories and Methods**

The design of each component and system considers the non-destructive examination (NDE) method that is used to fulfill Pre-Service Inspection and ISI examination requirements. Chapter 3, Subsection 3.10, describes requirements for in-service monitoring, testing, inspection, and maintenance programs for the BWRX-300. Chapter 3, Subsection 3.10.5, In-Service Inspection, states the requirements applied to components designated ASME BPVC Section III, Division 1 Class 1, 2 or 3 and their Subsection NF supports (Reference 5.11-3). In addition, the design considers the operational and radiological concerns associated with the selected examination method to ensure that the performance of the required examination is practical during normal plant operation. The 3D layout of the plant includes acceptance criteria regarding access for inspection equipment and personnel. The examinations detect flaws, surface and volumetric discontinuities/imperfections, and original component position discrepancies, including conditions such as:

- Cracks, wear, laps, seams, cold shuts, laminations, physical damage
- Corrosion type or erosion, debris
- Leakages
- Clearances, settings, physical displacements

- Loss of integrity at bolted or welded connections
- Missing or lost parts

NDE methods provide the best technique for inspecting components and equipment that have been disassembled or installed during maintenance.

#### **5.11.4.1 Reactor Pressure Vessel Welds**

Reactor pedestal wall and the RPV insulation mounted around the vessel inside the pedestal wall provide access for remotely operated Ultrasonic Testing (UT) examination devices.

#### **5.11.4.2 Reactor Pressure Vessel Head, Studs, Nuts, and Washers**

The RPV head, studs, nuts, and washers are dry stored on the refueling floor during refueling operations. RPV studs, nuts, and washers are cleaned, lubricated, inspected, and reused, if acceptable.

Replacement or repair of RPV studs, nuts, and washers is subject to additional Repair/Replacement Plan requirements per the ASME BPVC Section XI, Division 1, Article IWA-4000. The design provides for removable insulation to ensure access for manual UT examinations of the RPV head welds.

RPV studs are volumetrically examined in place or when removed.

RPV nuts and washers are accessible for visual examinations.

#### **5.11.4.3 Bottom Head Welds**

Access to the bottom head-to-shell weld is provided from the undervessel area through removable insulation around the bottom head, the CRD housings and nuclear instrument housings of the RPV. This design provides access for manual or automated UT examination equipment.

These welds are required to be accessible in order to perform ASME BPVC Section XI, Division 1, VT-2 examinations during system leakage testing in accordance with ASME BPVC Section XI, Division 1, Table IWB-2500-1.

Access to welds inside the bottom head is made through the respective housings during undervessel maintenance work. Access is provided for nozzle weld examinations.

#### **5.11.4.4 Reactor Vessel Support**

Access is provided for visual examination of the RPV support structure.

#### **5.11.4.5 Piping, Valves and Supports**

Design and physical arrangement of piping, valves, and supports provide personnel access to each weld location to perform NDE.

Working platforms facilitate servicing of valves and supports. Platforms and ladders provide access to piping welds, including the pipe-to-reactor vessel nozzle welds.

Removable thermal insulation is provided on welds and components, which require access for examination or are located within high radiation areas.

Welds are located to permit UT examination from at least one side.

A 100 percent volumetric in-service examination of all pipe welds is conducted during each inspection interval as defined within ASME BPVC Section XI, Division 1, Subarticle IWA-2400.

#### **5.11.4.6 Augmented In-Service Examinations**

##### ***5.11.4.6.1 High-Energy Piping***

NBS piping located between the CIVs, and extending to the Seismic Interface Restraint, is seamless (no longitudinal welds). After installation, circumferential welds in the aforementioned piping are subjected to 100 percent Radiographic Testing in accordance with ASME BPVC Section III, Division 1, Subparagraph NCD-5211.1 (Reference 5.11-4).

##### ***5.11.4.6.2 Flow Accelerated Corrosion Examinations***

ASME BPVC Section III, Division 1, Class 1, 2, and 3 piping systems, including piping and components, may be susceptible to FAC. This wear mechanism compromises the structural integrity (wall thinning) of high-energy carbon steel piping systems and components.

#### **5.11.4.7 Non-destructive Examination Methods**

NDE methods (surface, visual, and volumetric examinations) are described within ASME BPVC Section V (Reference 5.11-5), ASME BPVC Section XI, Division 1, and applicable codes and standards in Chapter 1, Appendix B.

For ASME BPVC Section III, Division 1, Class 1 and Class 2 welds, UT or Radiographic Testing examinations are selected for ISI examinations.

Radiographic Testing examinations are primarily used as a volumetric method. In addition, Radiographic Testing examinations supplement UT examinations to improve coverage of the required examination volume.

##### ***5.11.4.7.1 Surface Examination Methods***

Surface examinations of welds and bolts indicate the presence of surface discontinuities, which are performed by using Eddy Current Testing, Liquid Penetrant Testing, Magnetic Particle Testing, or UT methods.

Any linear flaw detected within a component by non-destructive techniques, which exceeds the allowable linear surface flaws standards is recorded.

##### ***5.11.4.7.2 Visual Examination Methods***

Visual examinations indicate the presence of surface discontinuities and imperfections, leakage from pressure-retaining components, and the general mechanical and structural condition of components.

##### ***5.11.4.7.3 Volumetric Examination Methods***

Volumetric examinations indicate the presence of discontinuities throughout the volume of the material, which are performed by using an Acoustic Emission (AE), Eddy Current Testing, Radiographic Testing, or UT method. Volumetric examinations are used for the examination of welds of vessels, piping, studs, and bolts.

#### **5.11.5 Inspection Intervals**

The ISI examination and system pressure test intervals, and percentage of examinations should conform with ASME BPVC Section XI, Division 1, Subarticles IWB-2400, IWC-2400, IWD-2400, IWF-2400, ASME OM Code, Subsections ISTC, ISTD, ISTF, and applicable codes and standards in Chapter 1, Appendix B. Inspection intervals are a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

MSRIVs and MSCIVs are periodically local leak rate tested (LLRT) to meet the leak rate acceptance criteria in accordance with the design requirements.

The MSRIVs and MSCIVs are diagnostically tested to verify that the MSRIVs and MSCIVs are properly adjusted to perform their respective safety MSL isolation functions, and to monitor for performance degradations.

MSL isolation trip channels are tested periodically to verify that MSRIV and MSCIV control pilots respond and position instrumentation is capable of providing required position signals to the respective control systems for required alarms and automatic trip functions.

MSRIV and MSCIV leakage rate testing is based on MSL cumulative allowable leakage rate established by DBA dose evaluation. Individual valve leak rate testing for diagnostic and rework/refurbishment is based on the design performance criteria.

FWRIVs leakage rate testing is based on total allowable leakage established by design analysis for PCS total leakage. Individual valve leak rate testing for diagnostic and rework/refurbishment is based on the design performance criteria.

NBS power-operated valves inside the RB are capable of remote manual functional testing. These valves are designed to be tested from the MCR. This testing verifies that proper position indication is displayed in the MCR.

#### **5.11.6 Provisions for Evaluating Examination Results**

Examination results are evaluated and accepted in accordance with ASME BPVC, Section XI, Division 1, Articles IWA-3000, IWB-3000, IWC-3000, IWD-3000, and IWF-3000, with repairs based on the requirements of IWA-4000, and applicable codes and standards listed in Chapter 1, Appendix B.

Recorded results meet the acceptance standards specified within ASME BPVC, Section XI, Division 1, Subarticles IWB-3400 and IWB-3500, IWC-3400 and IWC-3500, IWD-3400 and IWD-3500, IWF- 3400, and applicable codes and standards listed in Chapter 1, Appendix B.

Components containing flaws or relevant conditions and accepted for continued service are subjected to successive period examinations.

#### **5.11.7 System Pressure Tests**

NBS is designed with provisions for initial and periodic testing of the ASME BPVC Section III, Division 1, Class 1 and Class 2 equipment, including hydrostatic testing.

ASME system pressure tests (hydrostatic test, leakage test or pneumatic tests) are performed in accordance with ASME BPVC, Section XI, Division 1, Articles IWA-5000, IWB-5000, IWC-5000, IWD-5000, and applicable codes and standards listed in Chapter 1, Appendix B to ensure leak-tightness of components, equipment, mechanical and welded joints, integral support-piping attachment welds (i.e., lugs), and piping.

ASME system pressure-retaining boundaries are defined within ASME BPVC, Section XI, Division 1, Paragraphs IWA-5220, IWB-5222, IWC-5222, and IWD-5222.

VT-2 examinations are performed in accordance with ASME BPVC, Section XI, Division 1, Paragraphs IWA-2212, and applicable codes and standards listed in Chapter 1, Appendix B.

System hydrostatic test for RCPB systems is performed at a pressure and temperature, which corresponds to ASME BPVC, Section XI, Division 1, Subarticle IWB-5230, Table IWB-5230-1, ASME OM Code, Subsections ISTC, ISTF, and applicable codes and standards listed in Chapter 1, Appendix B.

#### **5.11.8 Surveillance Program Requirements and Overview**

The surveillance programs that are established address the several aspects of the materials used in the BWRX-300. Chapter 3, Section 3.10 provides additional information. Specifically, there is a need for:

1. Monitoring the integrity of the pressure-retaining RPV and its attachments
2. Monitoring the time limiting aging impact of the RPV
3. Monitoring of the integrity of the reactor internals
4. Monitoring the piping components for degradation
5. Monitoring the water chemistry program for its mitigation effectiveness
6. Monitoring for FAC

A discussion of the key elements of each surveillance program is presented in the next sections.

##### **5.11.8.1 Pressure-Retaining In-service Inspection Program**

There are ISI requirements based on the ASME BPVC, Section XI, Division 1 for pressure-retaining components and their integral attachments in the BWRX-300. This program serves as a model for the surveillance program that will include periodic visual, surface, and/or volumetric examination and leakage tests of the pressure-retaining components including the RPV, nozzles and attachments. The surveillance program will include specific scope for each of the pressure-retaining components. The program also will include defined preventative actions which include adequate methods to detect degradation that will occur with plant aging. The program will include defined monitoring and trending aspects, and specific correction action plans to address degradation identified. Finally, the program will track operating experience in these pressure-retaining components and use this information to update the ISI program.

##### **5.11.8.2 Reactor Pressure Vessel Material Surveillance Programs**

The material surveillance program monitors fracture toughness property changes of ferritic materials in the RPV beltline region resulting from exposure to neutron irradiation, including thermal information requirements pertaining to materials and surveillance capsules. The RPV material surveillance specimens are provided in accordance with requirements of the ASTM E185 standard and contain materials representative of the materials and material conditions used in the RPV beltline region. The plan accounts for the entire lifetime of the plant.

##### **5.11.8.3 BWRVIP-Type Reactor Internals Surveillance Inspections**

The reactor internal components are susceptible to aging degradation mechanism, particularly SCC and for the core shroud, and IASCC. While efforts are made to minimize degradation through material selection, component fabrication and the use of Hydrogen Water Chemistry/OLNC, the Boiling Water Reactor Vessel and Internals Project (BWRVIP) has developed inspection programs to monitor components for the impact of environmental degradation during plant operation in current operating BWRs in key internal components. The USNRC NUREG-1801, "Generic Aging Lessons Learned (GALL) Report" (Reference 5.11-6), provides an overview of these programs. These existing programs are used as the initial basis to develop BWRX-300 surveillance inspection plans for the different reactor internal components. The BWRVIP program includes inspection and flaw evaluation approaches that provide assurance of the long-term integrity. The guidelines provide information on component description and function, susceptible locations, and safety consequences of failure. They also provide recommendations for methods, extent, and frequency of inspection. The programs are focused on managing the effects of cracking due to SCC, IGSCC, or Irradiation Assisted Stress

Corrosion Cracking (IASCC), cracking due to fatigue, and loss of toughness due to neutron and thermal embrittlement. The BWRVIP program includes components such as the core shroud, core plate, core shroud support, top guide, core plate, steam dryer, and CRD housings.

#### **5.11.8.4 BWR Piping and Safe End Welds/HAZs Surveillance Inspections**

The BWRVIP following NRC guidance has developed inspection programs to monitor piping weld regions for the impact of environmental degradation, particularly IGSCC, during plant operation. These programs augment the ASME BPVC, Section XI, Division 1, ISI program to address IGSCC concerns. Chapter 5, Subsection 5.2.6 discusses the key measures that are taken to significantly reduce susceptibility of BWRX-300 piping components to SCC. The existing BWRVIP programs for ISI can be used as the initial basis to develop the BWRX-300 surveillance inspection plans for the different stainless steel piping welds/HAZ regions to minimize any SCC risks for degradation over the plant operating lifetime.

#### **5.11.8.5 Water Chemistry Monitoring/OLNC Effectiveness**

Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Revision 2 establishes that water chemistry is key to the BWR aging management plan for operating BWRs. This will also be the case for the BWRX-300. The BWRX-300 will comply with the BWR Water Chemistry Guidelines which address parameters for sample, frequency, and action levels. The guidelines provide proactive water chemistry guidance for mitigating environmentally assisted corrosion, maintaining fuel integrity, controlling FAC and controlling radiation fields.

#### **5.11.8.6 Flow Assisted Corrosion Monitoring**

For carbon steel piping, the FAC surveillance program defines key piping inspection locations, to evaluate and monitor wall thinning.

#### **5.11.9 Program and Milestone Implementation**

Refer to Chapter 13, Subsection 13.3.3.

#### **5.11.10 References**

- 5.11-1 ASME BPVC-XI-1," BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers.
- 5.11-2 ASME OM, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers.
- 5.11-3 ASME BPVC, "Section III, Division 1, Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers.
- 5.11-4 ASME BPVC, "Section III, Division 1, Subsection NF, Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers.
- 5.11-5 ASME BPVC, "Section V, Non-destructive Examination," American Society of Mechanical Engineers.
- 5.11-6 USNRC NUREG-1801, "Generic Aging Lessons Learned (GALL) Report.", Revision 2.

## **5.12 Reactor Auxiliary Systems**

### **5.12.1 Chemical and Inventory Control Systems for the Reactor Coolant**

IGSCC can occur in BWR startup and at-power environments. The normal BWR environment during power operation is 286°C water containing dissolved oxygen, hydrogen, and small concentrations of ionic and non-ionic impurities. Increasing levels of many ionic impurities influence SSC behavior of RCPB materials and may also affect fuel performance. Reactor water conductivity is maintained at or below the defined limits.

Besides being a major contributor to IGSCC of sensitized stainless steels, reduction of oxygen content is known to reduce the tendency for pitting and cracks of most plant materials. During power operation, most of the oxygen content of reactor water is due to the radiolysis of water in the core and, therefore, oxygen control cannot be achieved through traditional chemistry and operational practices. Reactor water oxygen control to low, plant-specific levels can be obtained through hydrogen injection from a Hydrogen Water Chemistry System. Hydrogen Water Chemistry is an established technique for mitigating and reducing the growth rates of IGSCC in reactor vessel internals. Control of reactor water oxygen during startup/hot standby is accomplished by utilizing the deaeration capabilities of the condenser. Refer to Chapter 9A, Subsection 9A.9.8 for further information.

In order to reduce the risk of IGSCC in reactor vessel internals, the BWRX-300 plant chemistry regime includes Hydrogen Water Chemistry System and On-Line NobleChem™ (OLNC). The OLNC operates in conjunction with the Hydrogen Water Chemistry System; although, they are both completely separate systems which have no mechanical or electronic interrelationships.

The OLNC provides a means for the injection of a noble metal salt solution directly into the reactor coolant flow path. In combination with Hydrogen Water Chemistry injection, the noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimize the oxygen available to initiate or encourage IGSCC crack growth. Based on the catalytic surface recombination efficiency provided by OLNC, less hydrogen is required to mitigate IGSCC initiation and reduce IGSCC crack growth rates. Due to the requirement for lower hydrogen, MSL dose rates are also lower.

The BWRX-300 water chemistry sampling and monitoring program is designed to analyze and monitor system chemistry for trending with alarm notification so actions can be taken to stay within operating specifications. The water chemistry control parameters, recommended operating limits, and recommended monitoring frequencies are developed to minimize the potential for IGSCC by controlling both ionic impurity and oxidizing radiolysis product concentrations in the reactor water. The Chemistry Control Program supports minimizing corrosion from chemical contaminants and monitoring chemical additives used to limit corrosion and contamination buildup. Chapter 13, Subsection 13.3.2.3 provides further details.

### **5.12.2 Reactor Water Cleanup System**

The Reactor Water Cleanup System (CUW) provides blowdown-type cleanup flow for the RPV during the reactor power operating mode. CUW also provides an overboarding flowpath (i.e., excess RPV coolant inventory discharge for level control) to the condenser hotwell or Liquid Waste Management System (LWM) directly from the RPV lower region.

During normal operating mode, the CUW system is in-service, with flow intake from the bottom of the RPV through two lines. The two inlet lines join into a single line outside the RPV where it exits the containment vessel, routes through piping in the steam tunnel and into the TB. Water flows through piping internally mounted inside the RPV downcomer region and exits the RPV greater than TAF+4m. At the top of this piping outside the RPV, there are double isolation valves integral



to the vessel, which close to prevent RPV inventory from exiting the CUW line in case of a pipe break.

CUW piping can be utilized to reduce reactor temperature stratification with reverse flow from the Shutdown Cooling System (SDC). To help reduce vessel stratification, the SDC water can be routed to the RPV lower region through the normal CUW inlet lines. This system operation is designed to provide flow to the bottom head region during startup to reduce thermal stratification caused by the continuous input of cold-water flow through the CRD seals.

### **5.12.3 High Point Vents of the Nuclear Boiler System**

The RPV head vent subsystem includes piping internal to the RPV head, two flange-mounted in-series RIVs, and piping to the MSL or Quench Tank, both located inside PCS. This vents non-condensable gases from the reactor steam dome via a connection to one of the MSL during plant operation. It permits the gases to be released from the RPV so that the RPV can be filled with water for hydrostatic testing and provides the upper tap for RPV level measurement during reactor shutdown. In addition, the horizontal portions of the head vent line piping to either the Quench Tank or the MSL have a downward slope in the direction of normal flow ensuring that no condensate stays in the line and prevents hydrogen from collecting.

A vacuum breaker is provided to equalize head vent line pressure with the PCS following cessation of flow through the vent line to the Quench Tank. The vacuum breaker is installed to prevent bypass leakage from the vent line to the PCS, or deposition of crud or debris in the vacuum breaker from the outlet side. The RPV head vent vacuum breaker is located at a high point near the downstream side of the Quench Tank pipe branch outboard isolation valve.

During reactor shutdown, reactor water level can be measured using the vent connection as the upper tap for the level instrumentation sensor. Also, during shutdown the NBS head vent connection permits air and non-condensable gases to be released from the RPV into the Quench Tank inside the PCS. The NBS head vent allows air to enter the RPV through the head vent connection when draining the RPV. During at-power operation, the NBS head vent connection permits and non-condensable gases to be released from the RPV into the MSL.

The non-condensable gases are extracted by differential pressure between the reactor head and the steam lines and then swept from the steam lines to the condenser.

### **5.12.4 Control Rod Drive System**

The CRD system contains components which form part of the reactor coolant pressure boundary, along with components which are important to safety to shut down the reactor. Those portions of the CRD system are classified as DL3/SC1.

The RPV provides integral CRD housing flanges for mounting the FMCRD to the vessel. The housing provides guidance and support for the FMCRDs in support of the normal control rod movement and scram functions. The RPV also provides mechanical interface at the coupling between the FMCRD and control rod, and at the coupling between the FMCRD and control rod guide tubes. These mechanical interfaces prevent ejection of a control rod in the event of a postulated failure of a CRD housing.

The FMCRDs provide electric-motor driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during off-normal conditions. The FMCRD major components that are part of the RCPB are analyzed and evaluated for the Service Level D conditions in accordance with ASME BPVC.

The CRD system materials specifications are discussed in Chapter 4, Section 4.5. The design of the CRD system is further described in Chapter 4, Subsection 4.6.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33955

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 6  
Engineered Safety Features**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CB	Control Building
CEPSS	Containment Equipment Piping Support Structure
CCS	Containment Cooling System
CFS	Condensate and Feedwater Heating System
CMon	Containment Monitoring Subsystem
CNSC	Canadian Nuclear Safety Commission
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CRH	Control Room Habitability
CRE	Control Room Envelope
CSA	CSA Group
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
DBA	Design Basis Accident
DEC	Design Extension Condition
DL	Defense Line
DPS	Diverse Protection System
DPT	Differential Pressure Transmitter
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
EFCV	Excess Flow Check Valve

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FPS	Fire Protection System
FW	Feedwater
GEH	GE Hitachi Nuclear Energy
HEPA	High Efficiency Particulate Air
HCU	Hydraulic Control Unit
HVS	Heating Ventilation and Cooling System
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Cooling & Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
LLRT	Local Leak Rate Testing
LOCA	Loss-of-Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
MSL	Main Steam Line
MSCIV	Main Steam Containment Isolation Valve
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
OLC	Operational Limits and Conditions
OPEX	Operating Experience
PCS	Primary Containment System
PCCS	Passive Containment Cooling System
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
POSAR	Pre-Operational Safety Analysis Report
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PSA	Probabilistic Safety Assessment
RB	Reactor Building
RBS	Reactor Building Structure
RCPB	Reactor Coolant Pressure Boundary
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SA	Severe Accident
SCCV	Steel-Plate Composite Containment Vessel
SC	Safety Class
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SSC	Structures, Systems, and Components
TAF	Top of Active Fuel
TGFU	Toxic Gas Filtration Unit
TRACG	Transient Reactor Analysis Code General Electric
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
WGC	Water, Gas, and Chemical Pads

## TABLE OF CONTENTS

<b>6.0</b>	<b>ENGINEERED SAFETY FEATURES .....</b>	<b>6-1</b>
6.1	Engineered Safety Feature Materials.....	6-1
6.1.1	Metallic Materials .....	6-2
6.1.2	Organic Materials.....	6-2
6.2	Isolation Condenser System (BWRX-300 Emergency Core Cooling System).....	6-3
6.2.1	Isolation Condenser System – General Description .....	6-4
6.2.2	Safety Design Bases .....	6-4
6.2.3	System and Equipment Function Description.....	6-6
6.2.4	Materials.....	6-9
6.2.5	Interfaces with Other Equipment or Systems .....	6-9
6.2.6	System and Equipment Operation .....	6-9
6.2.7	Instrumentation and Control.....	6-10
6.2.8	Monitoring, Inspection, Testing and Maintenance .....	6-12
6.2.9	Radiological Aspects .....	6-12
6.2.10	Performance and Safety Evaluation.....	6-13
6.2.11	References .....	6-13
6.3	Containment and Associated Systems .....	6-24
6.3.1	Containment Functional Requirements .....	6-24
6.3.2	Primary Containment System .....	6-25
6.3.3	Passive Containment Cooling System .....	6-40
6.3.4	Containment Isolation .....	6-44
6.3.5	Systems for Protection Against Overpressure and Underpressure .....	6-54
6.3.6	Filtered Venting System.....	6-54
6.3.7	Containment Leakage Testing .....	6-54
6.4	Control Room Habitability .....	6-58
6.4.1	System and Equipment Functions .....	6-59
6.4.2	Safety Design Bases .....	6-60
6.4.3	Control Room Descriptions .....	6-61
6.4.4	Materials.....	6-63
6.4.5	Interfaces with Other Equipment or Systems .....	6-63
6.4.6	System and Equipment Operation .....	6-63
6.4.7	Instrumentation and Control.....	6-64
6.4.8	Monitoring, Inspection, Testing and Maintenance .....	6-64



NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

6.4.9	Radiological Aspects .....	6-64
6.4.10	Performance and Safety Evaluation.....	6-65
6.4.11	References .....	6-65

**LIST OF TABLES**

Table 6.2-1: Structure, System, and Component Classification .....	6-14
Table 6.2-2: ICS System Interfaces.....	6-16
Table 6.3.2-1: Preliminary Primary Containment Key Design Parameters.....	6-33
Table 6.3.2-2: Primary Containment System Interfaces .....	6-34

## LIST OF FIGURES

Figure 6.2-1: Reactor Isolation Valve Assembly Example .....	6-19
Figure 6.2-2: Isolation Condenser System in Reactor Building .....	6-20
Figure 6.2-3: Isolation Condenser System Simplified Diagram .....	6-21
Figure 6.2-4: ICS Pools Simplified Flow Diagram (Pools) .....	6-22
Figure 6.2-5: ICS System Interfaces .....	6-23
Figure 6.3.2-1: General Containment Arrangement – Reactor Building Section View .....	6-36
Figure 6.3.2-2: General Arrangement – Reactor Building Section View .....	6-37
Figure 6.3.2-3: Primary Containment System Simplified Diagram (Containment Mechanical Penetrations) .....	6-38
Figure 6.3.2-4: Primary Containment System Interfaces .....	6-39
Figure 6.3.3-1: Passive Containment Cooling System .....	6-43
Figure 6.3.4-1: Penetration View Without Containment Wall .....	6-53
Figure 6.4-1: Main Control Room Habitability Envelope .....	6-66
Figure 6.4-2: Control Building HVAC Simplified Diagram .....	6-67
Figure 6.4-3: Control Building HVAC Simplified Diagram .....	6-68
Figure 6.4-4: Reactor Building Secondary Control Room .....	6-69

## **6.0 ENGINEERED SAFETY FEATURES**

The BWRX-300 incorporates Engineered Safety Features (ESFs) which mitigate the consequences of Anticipated Operational Occurrences (AOOs) or postulated Design Basis Accidents (DBAs) without any core damage. Information on the ESF materials is outlined in Section 6.1. Next, information on the Isolation Condenser System (ICS), which functions as the BWRX-300 Emergency Core Cooling System (ECCS), is captured in Section 6.2. The information on fission product containment and associated systems are presented in Section 6.3. Finally, the Control Room Habitability (CRH) function is outlined in Section 6.4. Supporting information is also described in Chapters 4, 5, 9A, 9B, 12, and 19.

ESF design features include passive systems that do not require dependence on external sources of power or operator actions fulfilling the Fundamental Safety Function (FSF). The BWRX-300 design does not require immediate operator responses for safety as reactor shutdown with passive vessel and containment heat removal are automatically initiated. ESFs mitigate the consequences of accidents that may cause major fuel damage and cannot be safely shut down by the Safety Class 1 (SC1) function alone. Features include the prevention of radioactive material release and plant personnel protection from radiation exposure. The engineered safety features are:

- Isolation Condenser System outlined in Section 6.2
- Containment and Associated Systems outlined in Section 6.3
- Control Room Habitability outlined in Section 6.4

The BWRX-300 design is inherently safe precluding the necessity for non-condensable gas control in the containment atmosphere following postulated DBA. Design Extension Conditions (DECs) or Beyond Design Basis Accidents (BDBAs) are described and analyzed in Chapter 15, Section 15.6. An overview of ESF materials selected in the BWRX-300 design is also included below.

### **6.1 Engineered Safety Feature Materials**

Materials used in the ESF components ensure that material interactions do not impair ESF operation. The selection of materials used in ESF functions is based on existing, historical Operating Experience (OPEX) for Boiling Water Reactors (BWRs). Quality standards are used for the design, fabrication, erection and testing of ESF components using applicable codes and standards for the safety class identified. Materials are selected to withstand the environmental conditions encountered during normal operation and postulated accidents, including Loss-of-Coolant Accidents (LOCAs). Material compatibility with core coolant water and the effects of radiolytic decomposition products are also evaluated.

The design, fabrication, erection and testing of the Reactor Coolant Pressure Boundary (RCPB) ensures a low probability of abnormal leakage or rapidly propagating failure or gross rupture.

Coatings used on exterior surfaces within the Primary Containment System (PCS) are suitable for expected environmental conditions.

### **6.1.1 Metallic Materials**

Materials selected in the design of the BWRX-300 Nuclear Boiler System (NBS) builds on a history of BWR material OPEX. RCPB materials are provided in Chapter 5, Section 5.2. Refer to Chapter 5, Table 5.2-1, *Typical Materials for Reactor Pressure Vessel Components* for the list of principal pressure-retaining materials and appropriate material specifications for the RCPB components. Refer to Chapter 5, Table 5.2-3, *Typical Reactor Internal Components and Materials* for the principal materials and associated specifications for the reactor internal components.

Details on material selection for ESF Structures, Systems, and Components (SSCs), including details on metallic and organic materials used in the ESF systems, are compatible for the intended use and described in the Pre-operational Safety Analysis Report (POSAR) in support of the Licence to Operate application.

All materials of construction used in ESF systems are resistant to corrosion, both in the medium contained and the external environment. General corrosion of all materials, except carbon and low alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel.

Demineralized water is employed in the Feedwater (FW) system. Refer to Chapter 10, Section 10.3 for a description of the water quality requirements. Based on OPEX, chloride leaching from concrete and other substances is not significant in BWRs.

No detrimental effects occur to any ESF materials from containment environment that affects the FW system. The materials are compatible with the post- LOCA environment.

### **6.1.2 Organic Materials**

ESF equipment materials are selected considering the effects of radiolytic and pyrolytic decomposition and attendant effects on safe operation of the system. Refer to Chapter 12, Section 12.3 for As Low As Reasonably Achievable (ALARA) design considerations.

Other organic materials in the containment are qualified to containment environmental conditions and are described in the POSAR.

#### **6.1.2.1 Evaluation of Materials for Engineered Safety Features**

Materials are specified to withstand expected radiation doses during the plant lifetime without any significant radiation-induced damage. The specified integrated radiation doses for design basis events are provided in the POSAR Chapter 15, Section 15.7.

Containment post-accident environment consists of hot water, nitrogen, and steam, no significant chemical material degradation is expected, due to strict inspection and testing. Solid debris from organic materials are unlikely.

## **6.2 Isolation Condenser System (BWRX-300 Emergency Core Cooling System)**

The BWRX-300 safety design philosophy for mitigating LOCAs credits conservative safety margins (e.g., larger water inventory), eliminates system challenges, and reduces the number and size of Reactor Pressure Vessel (RPV) nozzles relative to predecessor designs. In addition, all fluid system nozzles are located no less than four meters above the Top of Active Fuel (TAF). The large RPV volume, along with the tall chimney region, provides a substantial water reservoir above the core.

A Reactor Isolation Valve (RIV) assembly configuration is shown on Figure 6.2-1. These design features preserve reactor coolant inventory ensuring that adequate core cooling is maintained by isolation condensers removing the decay heat, following a LOCA.

The large RPV volume also reduces the rate that reactor pressurization occurs if the reactor is isolated from its normal heat sink. If isolation occurs due to a design basis event, the hydraulic scram function initiates reactor shut down and the Isolation Condenser System (ICS) functioning as the Emergency Core Cooling System (ECCS) removes reactor heat. The slower pressurization rate, hydraulic scram function, and ICS eliminate the need for relief and safety valves for pipe breaks and RPV isolation events. Refer to Chapter 15, Subsection 15.4.3 and Subsection 15.4.4 for further analysis of AOOs and DBAs.

The ICS consists of three independent trains, each containing a heat exchanger or Isolation Condenser (IC) that is submerged in a dedicated pool of water that provides the ultimate heat sink for protecting the reactor core for any off-normal event where the main condenser is not available, and the RPV is isolated.

The single smallest inner pool volume combined with the outer pool volume provides reactor decay heat removal for a minimum of three days. The inventory of the two smallest inner pool volumes, combined with the outer pool volume, supports reactor decay heat removal for a minimum of seven days. Each inner and outer ICS pool is connected through two in-series SC1, one-way flow devices that prevent backflow from the inner pools to the outer pools.

The heat rejection process is continued beyond seven days if the ICS inner pool inventory is replenished from an outside source. The ICS pools are located above ground and are not pressurized. Clean makeup water can be added directly to the ICS cubicles using diverse and readily available sources.

The Isolation Condenser Cooling & Cleanup System (ICC) (Chapter 9A, Section 9A.2.6) provides demineralized water makeup from the Water, Gas, and Chemical Pads (WGC) system (Chapter 9A, Section 9A.9) which compensates for the relatively minor evaporative water inventory losses occurring under normal reactor operating conditions.

ICS removes decay heat generated in the core and does not require coolant injection into the RPV to mitigate pipe breaks and transients. Refer to Chapter 15, Subsection 15.5.4 for small steam and liquid pipe breaks inside containment.

### 6.2.1 Isolation Condenser System – General Description

The ICS removes sensible and core decay heat from the reactor passively without any loss of reactor coolant inventory when the main condenser is unavailable. Reactor heat is transferred from each IC heat exchanger to the IC pool water by condensation and natural circulation. No forced circulation equipment is required when the ICS is in-service following these events:

- Reactor isolation at power operating conditions
- Loss of all alternating current (AC)
- Failure to Scram, (BDBA as described in Chapter 15, Section 15.6)
- LOCAs

Similar isolation condenser systems have been used in early BWRs (BWR/2 and BWR/3) as the emergency heat removal systems successfully for over 40 years. The genesis of the BWRX-300 IC is the Simplified Boiling Water Reactor IC design which was tested extensively for performance and other operating characteristics. The same IC design is specified in the Economic Simplified Boiling Water Reactor (ESBWR). The ESBWR IC heat transfer capacity was increased or uprated by the addition of rows of tubes within the original array. This power-uprated ESBWR IC version is used in the BWRX-300. Other modifications in the BWRX-300 IC design include design pressure (higher for BWRX-300), return into the chimney instead of the downcomer (refer to Section 6.2.6) and hardware material selections made to enhance the design.

### 6.2.2 Safety Design Bases

The BWRX-300 ICS is designed as a Defense Line (DL) 3 SC1 system that removes sensible and decay heat and provides RPV overpressure protection.

The ICS system meets the requirements of REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.5 (Reference 6.2-1). The technical justification and U.S. Nuclear Regulatory Commission (USNRC) acceptance of the ICS functioning as the ECCS is provided in NEDC-33911P-A, Rev. 3, *BWRX-300 Containment Performance* (Reference 6.2-2). ICS DL functions during normal and off-normal reactor plant operating conditions are organized by plant safety functions. While the reactor is at normal power operations, the ICS is in standby with the pressure boundary intact and ready to initiate on either a manual or automatic initiation signal.

The USNRC Safety Evaluation of NEDC-33910P-A, Rev. 2, *BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection* (Reference 6.2-3) justifies the ICS RPV overpressure protection for the American Society of Mechanical Engineers (ASME) Code Class 1 reactor pressure boundary. Refer to Chapter 15, Subsection 15.5.4 for limiting DBA overpressure events.

The single smallest inner pool volume combined with the outer pool volume provides reactor decay heat removal for a minimum of 72 hours without operator action. In plant modes where the Shutdown Cooling (SDC) is in-service, the ICS performs the SC1 DL3 function of isolating the interface connection(s) to the SDC in the event a pressure boundary leak is detected. This function ensures that passive failure or misalignment in a lower safety class system cannot disable the higher safety class ICS. The function is most critical when the RPV head is still tensioned, making it an important plant mode-related function.

ICS electrical and mechanical equipment is qualified in accordance with the BWRX-300 Equipment Qualification Specification using the established environmental conditions at the ICS pools and associated equipment locations so that they remain functional during post-accident environmental conditions, in addition to the normal conditions. Refer to Table 6.2-1 for the ICS SSC Classification.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

During normal plant operations, the ICS is in standby condition and performs the following safety functions:

- ICs and the condensate return lines are filled with water up to the steam distribution headers.
- Condensate return valves are closed, operable, and ready to open when ICS is initiated on a demand signal from interfacing DL2, DL3, and DL4a control systems.

Standby gas purge isolation valves are open venting gasses to the NBS main steam lines, operable, and ready to close when ICS is initiated on a demand signal from interfacing DL3 and DL4a control systems.

ICS pools at or above their minimum level and at or below their maximum temperature.

Check valves prevent backflow from each inner pool to the outer pool set are operable and preserve minimum inner pool inventory for DL2, DL3, and DL4a decay heat removal and RPV overpressure protection functions.

When SDC is not operating, interface connections to the lower safety class SDC system are isolated and perform their DL3 and DL4a reactor coolant pressure boundary integrity function.

During shutdown plant operation, the ICS provides a suction path from the RPV chimney internal region to SDC for the DL2 decay heat removal function.

In response to AOO or DBA events, ICS safety functions are confinement of radioactive materials (through overpressure protection and RCPB integrity), fuel cooling, and long-term heat removal. These functions are performed responding to various DL initiating events:

- ICS provides overpressure protection by maintaining RPV pressure below design limits. The overpressure protection function is performed in conjunction with the hydraulic scram function for overpressure event scenarios that progress to DL3. ICS overpressure protection is active in response:
  - a. RPV operating pressure setpoints exceeded (DL2)
  - b. RPV high pressure setpoints exceeded (DL3)
- ICS performs reactor coolant inventory addition and decay heat removal functions in response to:
  - a. RPV coolant level below setpoint (DL3 & DL4a)
  - b. Containment high pressure setpoint exceeded (DL3 & DL4a)
  - c. Line break indication (NBS main steam line, feedwater line, cleanup water, ICS) detected (DL3 & DL4a)
  - d. Main Steam Reactor Isolation Valve (MSRIV) or Main Steam Containment Isolation Valve (MSCIV) position out of range (DL4a)
- When SDC is in operation, ICS maintains the integrity of the RCPB in response to a FW or SDC line break by automatically isolating the interface connection to the affected SDC train (DL3 and DL4a).



### **6.2.3 System and Equipment Function Description**

The IC units located in the Reactor Building (RB) are submerged in an IC pool as shown in Figure 6.2-2 and simplified ICS system diagram in Figure 6.2-3. The ICs condense steam on the tube side and heat is transferred to the IC pool water which boils, and steam is vented to the atmosphere. The ICS pool arrangement provides the ultimate heat sink for protecting the reactor core for any AOO or DBA event where the main condenser is not available and the RPV is isolated. The IC is placed at an elevation above the steam source (RPV) so that this process is driven passively by gravitational force. When the steam is condensed, the condensate is returned to the RPV chimney via a condensate return pipe as shown in Figure 6.2-2. The steam side connection between the RPV and the IC is normally open, and the condensate return line is normally closed. This allows the IC and condensate return piping to fill with condensate that is maintained at a subcooled temperature by the ICS pool water during normal reactor operation. This is the standby mode or condition for the ICS.

The ICS is placed in operation by opening the condensate return line to the RPV. The subcooled condensate that is stored in the system during the standby state enters the RPV chimney interior providing additional inventory while quenching steam and lowering pressure at the reactor core exit. Simultaneously, steam from the RPV enters the IC where it is condensed in the tubes and returned to the RPV in a continuous cycle. If the RPV conditions fall below the saturation point, the ICS enters an idle state until decay heat drives conditions back to saturation, automatically and passively placing the ICS back into operation.

#### **6.2.3.1 Isolation Condenser System Pools**

The ICS pools consist of three main or inner pools and three expansion or outer pools. The inner pools are train specific, in that each only supplies cooling water inventory to the corresponding IC located within the pool compartment. Each IC pool is separated by a reinforced structural wall so that the ICs are not vulnerable to common cause failure. The ICS pools bidirectionally communicate with each other above their respective weir elevations through the outer pools. The weir level maintains pool separation. If a failure occurs in an outer pool, the inner pool level is maintained at no lower than the weir height, thus maintaining the inner pool volume supporting the system SC1 function.

The outer pools are cojoined above and below the weir elevation sharing cooling water with all inner pools and ICs. Each outer pool shares nomenclature with the adjacent inner pool even though the outer pools provide cooling water to all inner pools. The outer pools:

- Physically protect the inner pools from outside effects
- Provide water from all pools (inner and outer) to the ICC
- Provide water inventory for a minimum of seven days for ICS decay heat removal, even when assuming one ICS train is unavailable

Each outer ICS pool is connected to a cojoining inner ICS pool through doubly redundant, unidirectional, self-actuating flow devices that only permit flow from the outer pool to the inner pool, preventing inner pool inventory losses to a breached outer pool. The outer-to-inner pool one-way cross-connections are located within the pool compartments at an elevation that is below the IC tube vertical half-way point. This half-way point elevation defines the usable pools volume for boiloff heat removal capacity.

The simplified system diagram shown in Figure 6.2-4 illustrates the ICS pools arrangement. Note that "Inner Pool C" is the smallest of the three pools, as outlined in Figure 6.2-4. The minimum effective pool volume is down to the vertical half-way point of the IC tubes at an integrated reactor decay heat.

The ICS pools atmospheric vent piping originates from the ceiling of the ICS expansion pools and is routed vertically along the reactor building outside wall (refuel floor area) where it exits to the outside environment at an elevation below the polar bridge crane.

#### **6.2.3.2 Steam Supply, Condensate Return and Standby Purge Piping**

The steam supply, condensate return, and standby purge piping is designed to ASME Boiler and Pressure Vessel Code (BPVC) Section III, Subsection NB (Class 1) (Reference 6.2-4). The steam supply and condensate return piping extends from the outboard RIVs to the respective connection points with the associated IC subcomponents. The standby purge piping extends from the connection point with IC standby purge line subcomponent to the outlet of the outboard standby purge isolation valve.

The condensate return piping is routed from the connection point with the IC vertically or with downward slope in its entirety to the condensate return valves and loop seal. From the condensate return valves and loop seal, it is routed vertically connecting with the RIVs where any subcooled condensate forming at the steam to loop seal interface gravity drains to the RPV. The condensate return piping stored water volume is a critical parameter credited in the safety analyses, resulting in design constraints placed on the inside diameter piping and total length. The dynamic head losses or pipe friction losses of the piping are important to the safety analyses and are modeled in Transient Reactor Analysis Code General Electric (TRACG) analysis that forms the safety basis for the ICS design (Reference 6.2-5).

#### **6.2.3.3 Isolation Condenser Description**

The IC is designed to both ASME BPVC Section III, Subsection NB (Class 1) and ASME BPVC Section III, Subsection NCD (Class 2) (Reference 6.2-4 and 6.2-6). The steam supply and purge interconnection pipes are designed to Class 1 through the containment boundary up to and including the steam distributor and its integral flow restricting venturis. The purge line taps into the steam distributor and is routed back down through the annular area between the steam supply pipe and the penetration guard pipe to a connection point with the system standby purge piping. In effect, the purge line never traverses the containment boundary. The IC is designed to Class 2 downstream of steam distributor flow restricting venturis, including the upper header, condenser tubes, lower header, vent lines, and the two condensate drain lines up and not including the condensate tee and integral flow restricting orifices. The IC design returns to Class 1 starting with the condensate tee and includes the condensate piping, that routes back through the containment penetration to a connection point with the system condensate return piping.

Each ICS train has heat removal capacity greater than the reactor core decay heat at one minute after reactor shutdown with all control rods inserted. The known heat removal capacity of the ICs is established by full-scale prototype testing. The ICS core decay heat removal capability at one minute is outlined in Chapter 15, Section 15.5. IC heat removal capability increases when steam supply pressure increases, and it decreases when steam supply pressure decreases. The ICS trains have no common valves, piping, or RPV nozzles.

#### **6.2.3.4 Condensate Return Valves and Loop Seal**

Each ICS train has two 100 percent flow capacity condensate return valves installed in parallel. The two condensate return valves are remotely actuated with each designed to fail open upon a loss of control signal, control power, pneumatic supply, or hydraulic supply as applicable to the valve actuator design. One valve is designed for throttling while the other is a full-open/full-closed design. Both valves fail to the fully open position by stored spring energy. The valves are both located in horizontal runs of piping that are the lowest elevation of the condensate return piping, and the connection to the RIVs. This geometry creates a loop seal (discussed below) ensuring that the valves always have subcooled water on either side of their sealing surfaces, while the ICS is in standby, and while the reactor is in operation.

To place an ICS train in service, at least one of the two condensate return valves must open. The two valves installed in parallel ensure that no single active failure can cause the loss of any single train. The parallel valves and actuators are diversely designed, but both valves fail to the fully open position on a loss of pneumatics, control power, or control signal. Once the valves fully open, they stay fully opened until they are reclosed intentionally by the operator.

The water loop seal achieves two primary functions. First the loop seal performs a function under long-term operating conditions where the condensate return line liquid is low density and potentially in a two-phase state instead of a solid subcooled liquid column. The loop seal ensures a pocket of subcooled liquid exists in the condensate return line that prevents steam bypass conditions that would short circuit the system and lower heat removal performance. The vertical distance from the bottom of the loop seal to the RPV nozzle prevents steam bypass directly from the chimney region of the RPV to the isolation condenser by way of the condensate return line. Secondly, in the ICS standby condition, the loop seal ensures there is subcooled water on both sides of the condensate return valve seats that is nearly the same temperature. This eliminates valve body and seat distortion that could cause valve leakage.

#### **6.2.3.5 Standby Gas Purge Piping and Isolation Valves**

Whenever the ICS is in standby, and the reactor is critical, radiolytically-generated gases (hydrogen and oxygen) tend to accumulate in the upper point of the ICs where the steam to condensate interface exists within the steam distributor and associated piping subassembly. To prevent gas buildup eventually filling the distributor and the vertical steam supply line, a continuous gas purge is required. The purge line taps into the center region of the steam distributor and is routed back down into the Steel-Plate Composite Containment Vessel (SCCV) inside the penetration space between the vertical steam supply pipe and the outer guard pipe. The purge line is routed to the Main Steam Line (MSL) at a location between the RIV and the containment penetration.

The purge lines for ICS trains A and B connect to MSL A and the ICS train C purge line connects to MSL B. The pressure drop that occurs between the interior of the RPV and downstream of the RIV induces purge gas flow from the IC steam distributor to the MSL. Inside the SCCV, the gas purge line contains two automatic isolation valves installed in series that fail close on a loss of signal or power. The two in-series valve design provides redundancy in the event of a single active failure of one valve to close on demand. At least one of these valves close whenever the reactor is isolated, and the ICS is placed into service. The valve(s) close to prevent loss of inventory when the RPV is isolated.

The IC has an integrally mounted catalytic recombiner device that processes radiolytic gases when the system is in operation, eliminating the need to vent non-condensable gas. The catalytic recombiner does not alter the IC heat transfer characteristics.

#### **6.2.4 Materials**

ICS electrical and mechanical equipment is qualified using environmental conditions where they are located. ICS components functioning under faulted conditions remain functional under the post-accident environmental conditions in addition to the normal conditions. The IC is a full-scaled prototype tested component. The material specifications for system and subcomponents design pressures/temperatures are provided in the POSAR. The seismic integrity of the ICS pools is discussed in Chapter 9, Section 9B.2.3.

#### **6.2.5 Interfaces with Other Equipment or Systems**

Figure 6.2-5 identifies the ICS system interfaces and Table 6.2-2 summarizes the system interfaces. Each ICS train directly interfaces with RPV, and the Boron Injection System (BIS) or SDC.

##### **6.2.5.1 Interface Connections with Shutdown Cooling**

ICS trains A and B each have two series isolation valves outside the containment penetrations in the interface lines to the SDC trains A and B (refer to Chapter 9A, Section 9A.2.3) that provides a SDC suction path from the RPV chimney interior. The interface piping in the ICS is a connection located in the loop seal region between the condensate return valves and the RIVs. The interface piping is routed through a containment penetration and includes two in-series outboard remote-actuated isolation valves. The actual ICS to SDC boundary occurs at the outlet of the most outboard isolation valve for each train.

The double isolation in-series valve design provides redundancy in the event of a single active failure to one valve. The isolation valve closest to the SCCV penetration has two safety class functions. One function is normal containment isolation where the valve works in tandem with its associated RIV. The other function is terminating inventory loss in the event of a SDC pressure boundary failure outside containment when the plant is shut down and relies on the SDC for decay heat removal and cooldown. For this function, the two outside isolation valves work in tandem, providing redundancy in the event of a single active failure. Even though the RIV and outside containment isolation valve can provide the same redundant protection against a SDC breach, the ICS train associated with that interface would not be functional for decay heat removal. The availability of a functional ICS train is especially critical while the RPV head is still installed or prior to flood up.

Refer to Subsection 6.3.4 for containment isolation functions of the SDC interface valves.

##### **6.2.5.2 Interface Connection with Boron Injection System**

ICS train C interfaces with the BIS in a similar fashion as the SDC. The system boundary isolation valves are part of the BIS, and the system interface occurs inside the containment building (refer to Appendix 15B). The BIS system has one remote isolation valve outside containment and one check valve inside containment. The BIS system interfaces with the check valve inboard side.

#### **6.2.6 System and Equipment Operation**

Upon receipt of a RIV actuation, at least one condensate return valve in one ICS train opens and at least one standby gas purge isolation valve in each train closes. The subcooled condensate stored in the system during the standby state enters the RPV chimney interior providing additional inventory while quenching steam and lowering pressure at the reactor core exit. Simultaneously, steam from the RPV enters the IC where it condenses in the tubes and returns to the RPV in a continuous cycle. If the RPV conditions fall below the saturation point, the ICS enters an idle state until decay heat drives conditions back to saturation, automatically and passively placing the ICS back into operation.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

With ICS in operation (one, two or three trains as required), decay heat removal does not exceed the plant Operational Limits and Conditions (OLCs) for RPV cooldown rate. One ICS condensate return valves in each train is used to throttle flow to maintain RPV pressure, producing the desired RPV cooldown rate. Eventually, one ICS train is sufficient to remove reactor decay heat. When the ICS is operating, the purge isolation valves close preventing the loss of RPV inventory. Radiolytic hydrogen and oxygen are recombined with passive catalytic recombiners installed integral to each of the ICs.

ICS control system functions are described in Subsection 6.2.7.

Because there is no significant oxidation, the only non-condensable gases that may migrate into the IC tubes are the radiolysis products following a design basis LOCA. The build-up of hydrogen and oxygen in steam following a 19 mm liquid break is analyzed using TRACG (Reference 6.2-5). TRACG shows that the radiolytic gas build up in the IC does not cause a significant degradation in the IC performance because any potential buildup occurs downstream of the heat exchanger tubes in the lower drum where the radiolytic gases are removed by catalytic recombination. Furthermore, the sensitivity calculations demonstrate that the effects of the potential degradation in the IC performance resulting from the presence of radiolytic gases are small even for much higher than reasonably expected radiolytic gas concentrations. The design limits the radiolytic gas volume fraction in the IC lower drum to less than 4%.

Unlike hydrogen generated from cladding oxidation, hydrogen and oxygen liberate from radiolysis at a slow rate and distribute over a relatively large region. The gas becomes mixed in water both in steam and liquid phases. The well-mixed hydrogen and oxygen in steam also migrates into the ICs when the ICs are in-service. The hydrogen and oxygen concentration in steam may increase as the steam condenses in the IC tubes. Because hydrogen and oxygen are mixed in water when they are formed near the core region, they do not separate from water or steam again until the steam condenses.

Since the ICS is a closed system attached to a high purity reactor coolant system, it is not subject to flow blockages, particularly since the steam intake is from a high elevation in the RPV and the condensate discharge is to an elevation above the TAF.

An ICS train can only be isolated automatically if a detectable leak or break occurs.

### **6.2.7 Instrumentation and Control**

The ICS provides emergency core cooling and is also the primary system preventing RPV overpressure. Diversity, redundancy, and backup power is supplied for the instrumentation and controls ensuring ICS operation when initiation occurs.

#### **6.2.7.1 Steam Condensate Temperature Instrumentation**

Each ICS train has temperature instrumentation installed on the steam supply line and condensate return lines. The temperature instrumentation installed on the steam supply lines is located as close to the containment penetration as practicable. The temperature instrumentation location allows for determining whether the steam line is sufficiently heated during startup and low power reactor operation.

Each IC train has temperature instrumentation installed on the upper and lower headers that measures the temperature of the bulk fluid. Temperature measurement of the lower header provides confirmation that the catalytic recombiner is functional and prevents buildup of non-condensables when the ICS is in operation.

#### **6.2.7.2 Isolation Condenser System Pool Level and Temperature**

Each ICS inner pool is equipped with narrow range level instrumentation at the weir elevation. Each ICS inner pool is equipped with level instrumentation that measures the wide range pool level while the ICS is in operation.

Each ICS inner pool is equipped with bulk pool temperature instrumentation.

#### **6.2.7.3 RPV Isolation from ICS Following an ICS Line Break**

The safety design basis assumes an ICS train failure occurs at the pressure boundary that necessitates quick closure of the RIVs for that train steam supply and condensate return lines to terminate a LOCA. In this event, the other two ICS trains remain intact and are not inadvertently isolated to mitigate the event (cool and depressurize the RPV). To ensure that inadvertent isolation of an ICS train does not occur at any time, two important features exist in the physical design and associated logic:

- The RIVs that interface with ICS are designed to fail as-is and are open under normal operating modes. This contrasts with all other RIVs, which have a fail-closed design.
- Redundant and diverse instrumentation and controls are provided.

The ICS RIVs closing function is described in Chapter 7, Section 7.3.

Each ICS train is equipped with two flow detection elements in the form of 90-degree bends that have impulse lines with three parallel Differential Pressure Transmitters (DPTs). One flow detection device is located in the steam supply line, and another located in the condensate return line. These DPTs detect flow within the system. If the ICS train is in standby and flow is detected, a line break is indicated and the control system for this line initiates RIV closure for this train only. ICS leak detection is not active during ICS operation, because the heat removal FSF is prioritized over LOCA mitigation and no automatic closure of RIV occurs. An ICS line leak occurring during ECCS FSF is a BDBA.

#### **6.2.7.4 System Actuation, Opening the Condensate Return Valves and Closing the Gas Purge Isolation Valves**

As indicated in Subsection 6.2.6, when an ICS initiation event occurs, two initiations occur that place the ICS into alignment for event mitigation:

- At least one condensate return valve in one ICS train opens
- At least one standby gas purge isolation valve in each train closes.

The number of ICS trains placed into service depends on the event severity as a result of RPV pressurization rate. In the most severe event, condensate return valves in each train may open on a staggered basis. The parallel redundancy of the condensate return valves and the series redundancy of the standby gas purge isolation valves ensure that any single active failure does not disable any of the ICS trains.

The ICS operational logic places into service only the number of train(s) required to limit an RPV pressure increase and then holds pressure stable at a predetermined value, or lowers pressure at a prescribed rate, depending on the event circumstances. Both the modulating and full-open demand setpoints are staggered across all condensate return valves by train. The established setpoints modulate valves in all three trains initially limiting and then controlling RPV pressure with each train's setpoint incrementally higher than the lowest. Controlling RPV pressure is a DL2 function and is controlled by the SC3 Isolation Condenser Throttling Pressure Controller (refer to Chapter 7, Subsection 7.3.3).

Should the RPV pressure fail to stabilize or continue increasing, setpoints are reached that trigger both condensate return valves in each train to fully open in an incrementally staggered sequence by train. Refer to Chapter 7, Table 7.3-1, *C10 SC1 I&C Functions and Initiation Signals*. The full open condensate return valve is a DL3 function (full open or close only valve) and a DL4a function controlled by the diverse protection system (DPS) control platform (both full open/close and modulating valves).

### **6.2.8 Monitoring, Inspection, Testing and Maintenance**

Pre-operational and in-service testing for the ICS include:

- Instrumentation calibration
- Instrumentation and equipment operation combinations using logic and instrument channel trip
- Instrumentation and alarm monitoring functionality and availability
- Valve operation and timing
- Verification that steam flow paths from the ICS pools to atmosphere are unobstructed
- Verification that IC steam and condensate return piping flow passages are unobstructed
- IC pool level control operation
- Verification that IC pool sub-compartment valves are locked open
- IC containment isolation valves close upon simulated isolation signal receipt
- Instrument channel response times measured from each process variable input signal to the process actuator confirmation signal
- Periodic stroke testing of each normally closed condensate return valve during normal plant operations with the RIVs closed preventing unnecessary system piping and components thermal cycling
- Periodic stroke testing of each normally open standby IC gas purge valves during normal plant operations

Periodic heat removal testing of each IC is performed on a rotating basis at an operating cycle interval.

The ICS is an SC1, Quality Group A and B system that is designed and constructed as ASME BPVC Class 1 and 2 (Reference 6.2-4 and 6.2-6). OLC surveillance tests include in-service component tests and in-service pressure boundary component inspections (piping welds). The design and construction of the ICS integrates features that prevent thermal fatigue and permit tests and inspections.

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets. Refer to Chapter 16 for a discussion on OLCs.

### **6.2.9 Radiological Aspects**

Large ICS breaks outside containment and small ICS line breaks are discussed in Chapter 15, Subsection 15.5.9.

The mass and energy release from the isolation condenser breaks outside containment are bounded by the isolation condenser pipe breaks inside containment that are described in Chapter 15, Subsection 15.5.4. The isolation condenser pipe breaks do not use any DL3 functions that depend on containment parameters and the containment back pressure is not credited in any isolation condenser pipe break analyses.

As is the case for isolation condenser pipe breaks inside containment, core response is not a concern since the break is isolated rapidly. The consequences of large isolation condenser pipe breaks outside containment are evaluated for loads, pressures, and temperatures outside containment and radiological consequences resulting from normal operation coolant activity.

Since the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment. The ICS break analyzed for offsite dose consequences is a postulated break of an IC steam side component within the ICS pool compartment of the RB. The break is immediately detected and RIV isolation occurs in 10 seconds. Refer to Chapter 15, Subsection 15.5.9 for ICS. As shown in Chapter 15, Table 15.7.7-3, *BWRX-300 DNNP Isolation Condenser System Line Break Outside Containment Accident Dose Consequences*, the dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin.

#### **6.2.10 Performance and Safety Evaluation**

In general, any one ICS train can mitigate AOOs. Two ICS trains are required for LOCA mitigation (analysis assumes one of the three ICS trains has a single failure). All three ICS trains are credited for beyond design basis events and the IC pool inventories can be replenished indefinitely. Functions of the ICS satisfy the analysis assumptions for AOOs, and DBAs as described in Chapter 15, Section 15.5. For BDBAs, refer to Chapter 15, Section 15.6.

#### **6.2.11 References**

- 6.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 6.2-2 NEDC-33911P-A, "BWRX-300 Containment Performance," GE-Hitachi Nuclear Energy Americas, LLC.
- 6.2-3 NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," GE-Hitachi Nuclear Energy Americas, LLC.
- 6.2-4 ASME BPVC, "Section III, "Rules for Construction of Nuclear Facility Components," Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 6.2-5 NEDC-33987P, "BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application," GE-Hitachi Nuclear Energy Americas, LLC.
- 6.2-6 ASME BPVC, "Section III- Rules for Construction of Nuclear Power Plant Components," Subsection NCD: Class 2 Components," American Society of Mechanical Engineers.



NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 6.2-1: Structure, System, and Component Classification**

Principal Components	Safety Class	Location	Quality Group	Seismic Category
Steam supply, condensate return, standby gas purge piping	SC1	CV	A	A
SDC interface piping to containment isolation valve, A and B trains	SC1	CV, RB	A	A
BIS interface piping to BIS interface valve, C train	SC1	CV	A	A
SDC interface piping from containment isolation valve to downstream redundant isolation valve, A and B trains	SC1	CV	B	A
ICS pools atmospheric vent piping	SC1	RB	B	A
Outer pool to inner pool cross-connect piping	SC1	RB	B	A
Long term ICS pool makeup piping (also referred to as flex-makeup piping)	SC3	RB	D	A
Flow detection differential pressure instrumentation impulse piping	SC1	CV, RB	B	A
Isolation Condensers <sup>(1)</sup>	SC1	CV, RB	A, B	A
Condensate return valves (Open/Closed Only)	SC1	CV	A	B
Condensate return valves (Throttling)	SC2	CV	A	B
Standby gas purge valves	SC1	CV	A	B
Containment isolation valve to SDC system, A and B trains	SC1	CV	A	B
Outer pool to inner pool cross-connect backflow prevention devices	SC1	RB	B	B
Redundant isolation valve to SDC system, A and B trains	SC1	CV	B	B
Flow detection differential pressure instrumentation, including tubing, cables, trays, and instrument racks	SC1	RB, CV	—	A and B
Wide range pool level instrumentation used for post-accident monitoring, long term (>72 hours)	SC1	RB	—	B
All piping installed temperature instrumentation, pool temperature instrumentation and narrow range pool level instrumentation used	SC3	RB, CV	—	Non-Nuclear Seismic

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

Principal Components	Safety Class	Location	Quality Group	Seismic Category
Pneumatic supply tubing and components from the actuator to the control solenoid valves for the open/closed only condensate return valves, containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, CV	—	A
Hydraulic supply tubing and components from the actuator to the control solenoids valves for the throttling condensate return valves	SC2	CV	—	A
Control solenoid valves for the condensate return valves, containment isolation valves, and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, CV	—	B
Control solenoid valves for the throttling condensate return valves	SC2	CV	—	B
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System to the control solenoids valves for the condensate return valves	SCN	RB, CV	—	Non-Nuclear Seismic
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System (not including the check valve) to the control solenoids valves for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	—	A
Hydraulic supply tubing from the positioner to the control solenoid valves for the throttling condensate return valves	SC3	CV	—	A
Pneumatic supply check valve to the valve actuators for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	—	B

- (1) The steam supply components from the steam supply piping to and including the flow restrictors, and the condensate return components from and including the 6"X8" tee to the condensate return piping are Quality Group A. All components between the steam flow restrictors and the condensate tee including headers, drums, tubes, flanges, and covers are Quality Group B.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 6.2-2: ICS System Interfaces**

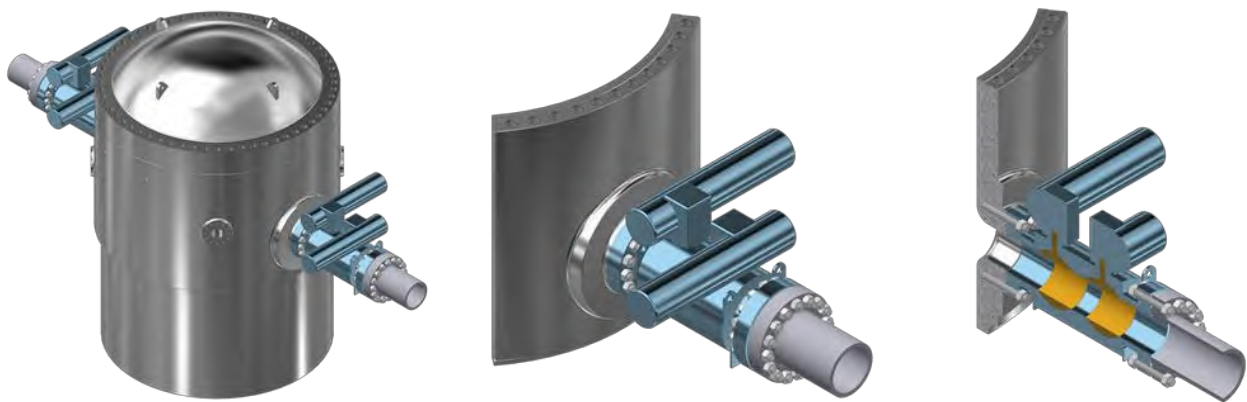
<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	Decay heat removal, receive steam, return condensate. Accept standby mode IC gas purge.	RIVs, steam supply and condensate return RPV nozzles. Gas purge to main steam lines at a location between the RIVs and the containment penetration.
Safety Class 1 Instrumentation and Control	Safety Class 1 Instrumentation and Control System to receive/provide signals to Safety Class 1 instrumentation and controls in ICS.	Interface boundary is at the input/output terminations in the Distributed Control and Instrumentation System cabinets within the scope of the SC1 I&C control equipment.
Safety Class 2 and 3 Instrumentation and Control	Safety Class 2 and 3 Instrumentation and Control System to receive/provide signals to Safety Class 2 and 3 instrumentation and controls in ICS.	Interface boundary is at the remote multiplexer unit associated with the instrument or control equipment.
Process Radiation and Environmental Monitoring System	Monitor ICS pool compartments for elevated radiation levels that may indicate ICS pressure boundary leakage.	Radiation detectors mounted in the ICS pool compartments or atmospheric vent line.
Boron Injection System	<p>ICS serves as a boron injection point via the train C condensate return line to the RPV chimney interior.</p> <p>The ICS condensate return piping for train C provides the interface between the BIS and the RPV. This interface avoids the need dedicated RPV penetration for BIS. The ICS provides the interface piping that extends from the condensate return line to a system boundary isolation valve that is in the scope of the BIS and is located inside the containment vessel.</p> <p>The ICS interface piping to the BIS connects to the condensate return piping between the condensate return valves and the RIVs at the lowest horizontal section of piping that includes the return isolation valves, effectively in the zone of the loop seal. This ensures that all of the interface piping to the BIS contains subcooled water.</p>	Pipe connection on the ICS train C condensate return line at a location between the condensate return valves and the RIVs.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
ICS Pool Cooling and Clean up System	Provide ICS pool water continuous cleanup to meet water quality standards and maintains the pool water below the temperature limit. Also, provides normal operational makeup water to maintain pool water levels.	The interface occurs through seismically qualified suction and discharge connections to the ICS pools.
Shutdown Cooling System	<p>ICS serves as a Shutdown Cooling suction point via the trains A and B condensate return lines from the RPV chimney interior.</p> <p>The ICS condensate return piping for trains A and B provides the interface between the SDC system and the RPV. This interface avoids the need for two RPV penetrations dedicated to shutdown cooling. The ICS provides the interface piping that extends from the condensate return line and out through the containment vessel where the connections to SDC occur. There are two ICS scope remote-actuated isolation valves outside of containment in each interface line. (Requirements pertaining to the outside containment isolation valves are located in Chapter 5, Section 5.10.) The SDC system begins at the outlet of the most outboard valve. This arrangement maintains the SC1 and SC2 piping and components within the scope of the ICS.</p> <p>The ICS interface piping to the SDC system connects the condensate return piping between the condensate return valves and the RIVs in the lowest horizontal section of piping that includes the return isolation valves, effectively in the zone of the loop seal. This ensures that all of the interface piping to SDC contains subcooled water.</p>	Pipe connection on the ICS trains A and B condensate return lines at a location between the condensate return valves and the RIVs.
Liquid Waste Management System (LWM)	Provide a storage vessel for ICS pool water that may need to be relocated to facilitate component maintenance or inspection that is located within the pool compartments, such as the ICs.	At this time the interface is made by temporarily installed pump taking suction from the refueling floor elevation and discharging through a flexible connection to installed piping that is in the scope of LWM system.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Plant Pneumatics System	<p>Provide pressurized nitrogen gas to charge accumulators for the condensate return valve actuators and the SDC interface isolation valve actuators.</p> <p>The function of supplying nitrogen to the condensate return valves is non-safety class. The valves are held closed for the standby mode by nitrogen pressure, so the plant pneumatics function is important to prevent inadvertent actuation of the system. An accumulator and its inlet check valve, which are in the scope of ICS, store an adequate supply of nitrogen as a coping measure in the event that the plant pneumatics nitrogen source is lost. Since the condensate return valves fail to the open or safe position on loss of pneumatics, the nitrogen supply piping, accumulator, and check valve are not SC components</p>	Interface boundary is at the inlet of the Safety Class N backflow preventors.
Safety Class 1 Electrical Distribution System	Safety Class 1 Electrical Distribution System and batteries provide power to Safety Class 1 electrical loads in the ICS system.	Interface boundary is at the input to the local breaker at the Motor Control Center termination.
Primary Containment System	Provide the piping penetration sleeves and connections to ICS system piping ensuring a leak tight barrier.	Interface occurs at the penetration sleeve to piping weld.
Reactor Building Structure (RBS)	Provides ICS pool compartments and ICS support structure overall support (e.g., IC supports, piping supports)	Interface occurs at the ICS component support to RB structure attachment points (e.g., fasteners, welds).



**Figure 6.2-1: Reactor Isolation Valve Assembly Example**

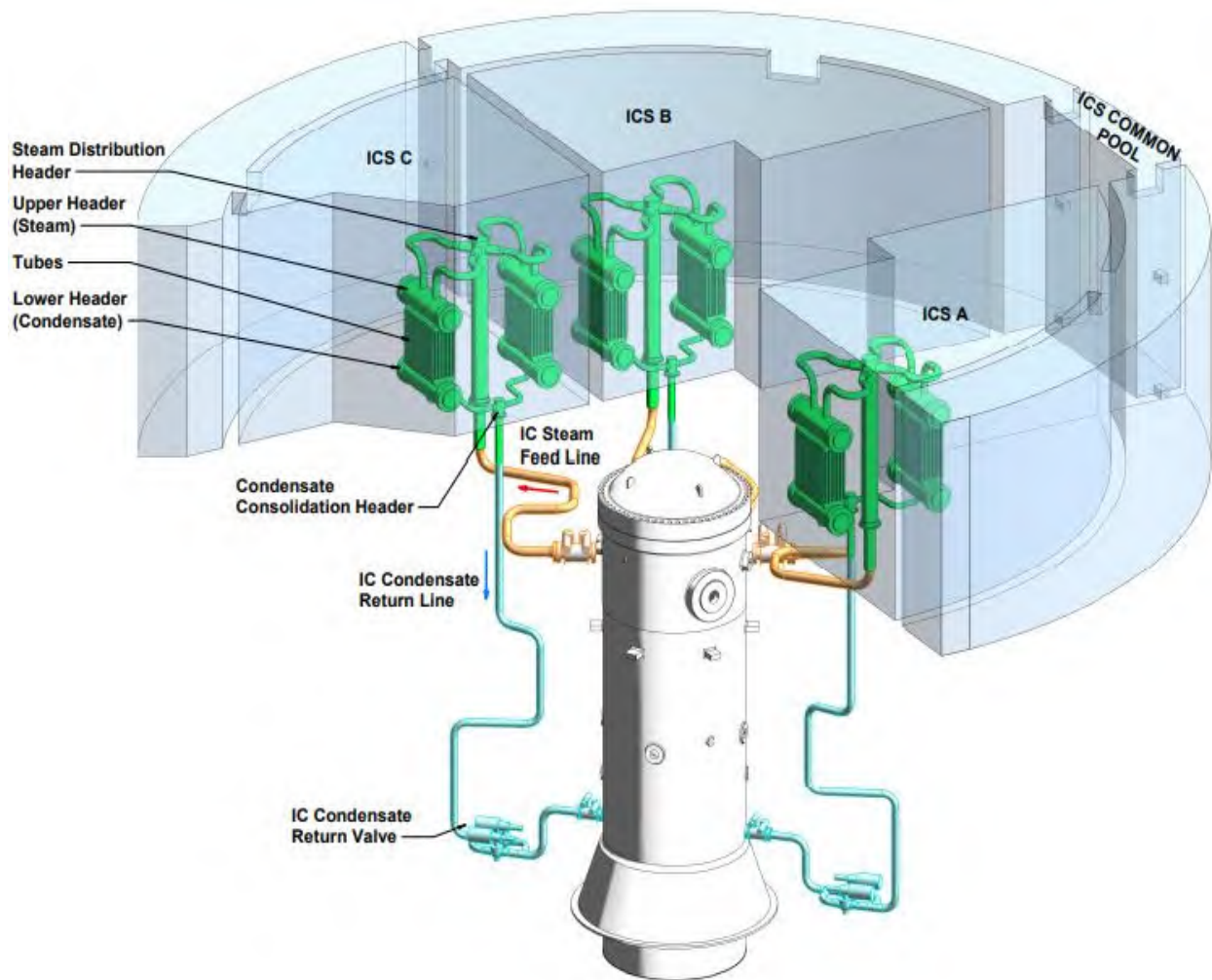


Figure 6.2-2: Isolation Condenser System in Reactor Building

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

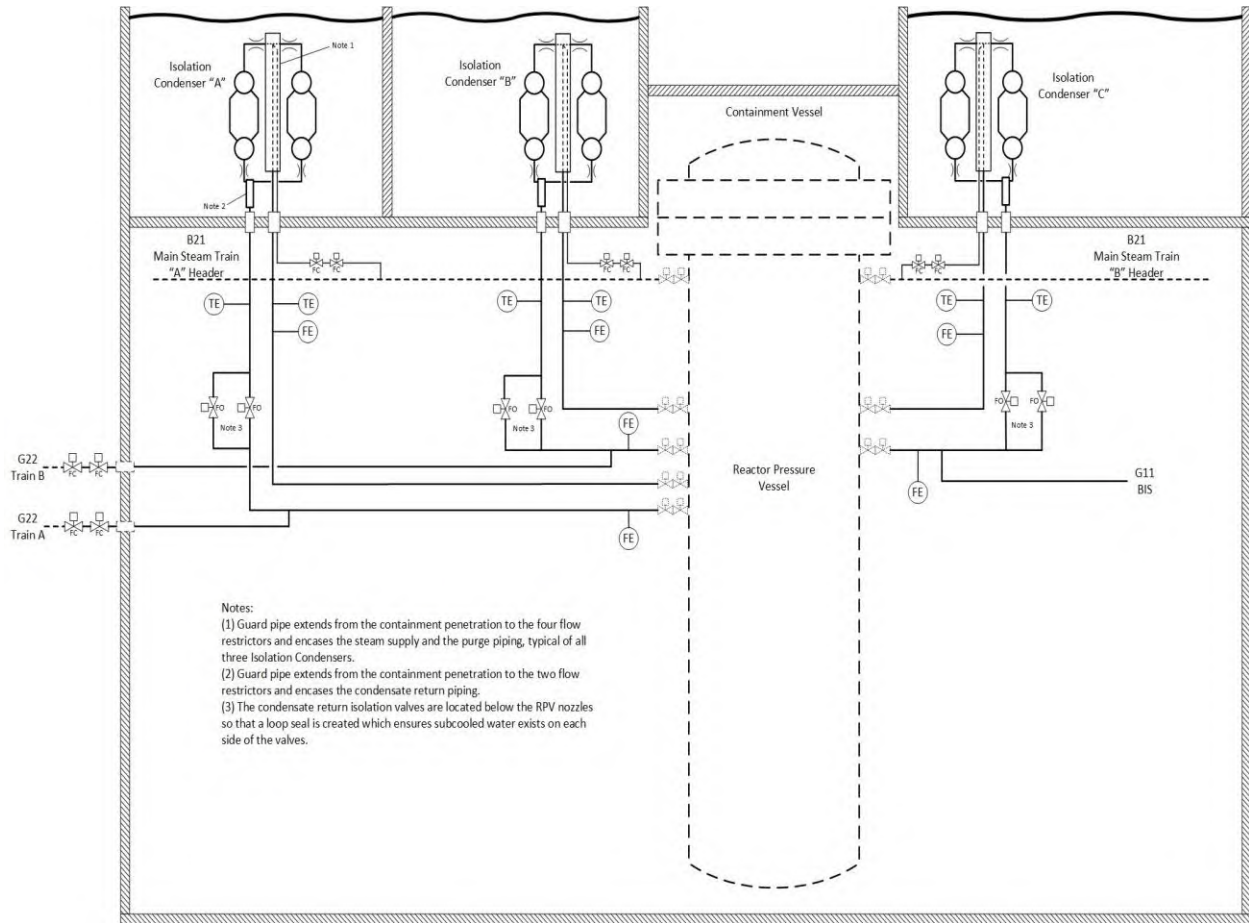
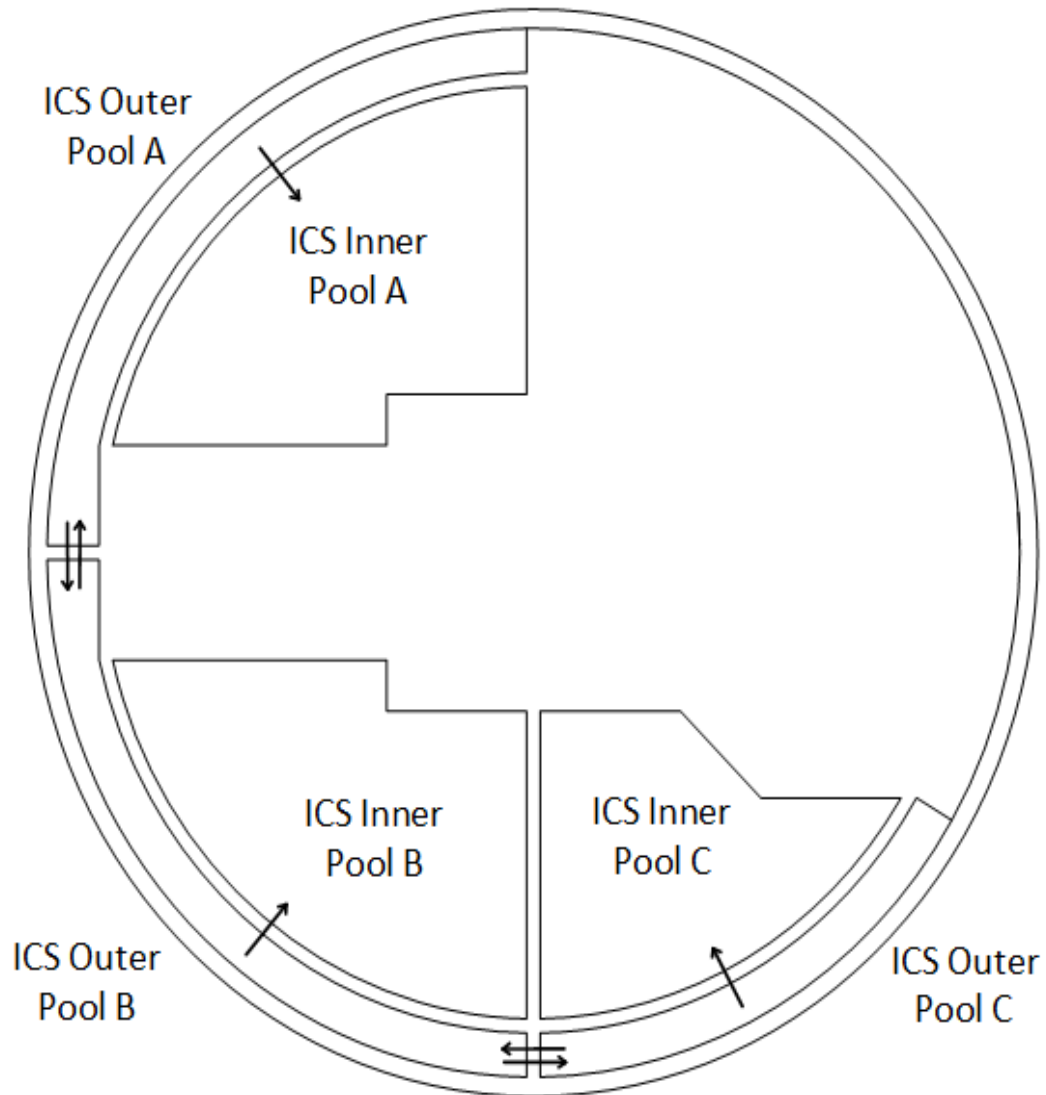


Figure 6.2-3: Isolation Condenser System Simplified Diagram





**Figure 6.2-4: ICS Pools Simplified Flow Diagram (Pools)**

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

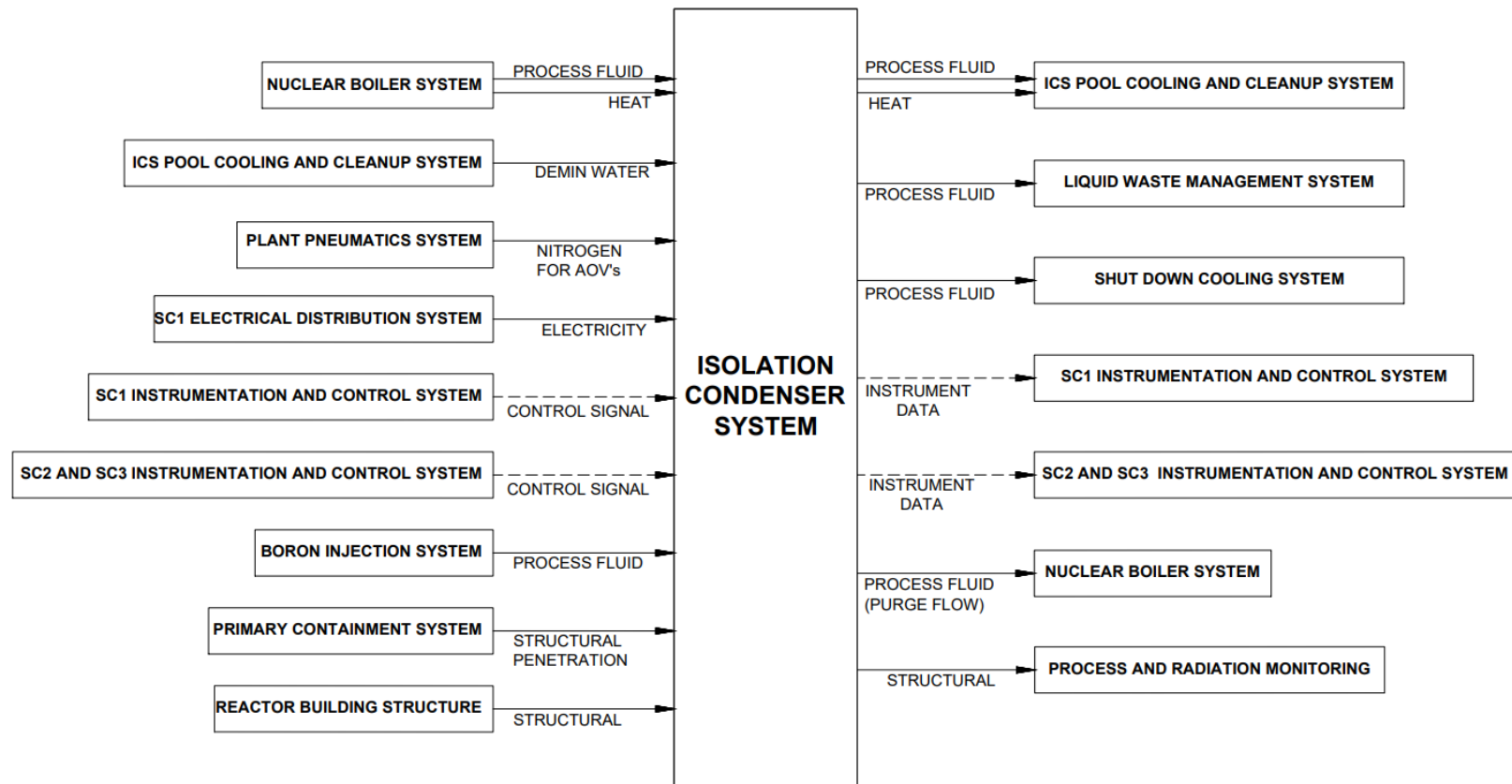


Figure 6.2-5: ICS System Interfaces

### **6.3 Containment and Associated Systems**

The BWRX-300 containment pressure boundary provides a SC1 leak-tight barrier to prevent the release of radioactive material to the environment in the event of a failure of the RCPB. The containment structure maintains its functional integrity during and following peak transient pressures and temperatures that could occur following postulated DBAs. The containment integrity is assured by the isolation of mechanical systems penetrating containment for Postulated Initiating Events (PIEs) that could result in an uncontrolled release of radioactive materials to the environment. Other containment systems support the passive removal of heat generated from AOOs, DBAs, and DECAs, and the control of hydrogen and other combustible gases from Severe Accident (SA) or BDBAs.

#### **6.3.1 Containment Functional Requirements**

The PCS design limits are established and confirmed by Deterministic Safety Analysis (DSA) for postulated AOOs and DBAs (refer to Chapter 15, Section 15.5). The PCS provides the FSF of confinement that achieves the following confinement functions:

- Control of pressure and temperature
- Isolation of the confinement boundary
- Leak-tightness of the confinement boundary
- Controlled point of release
- Control of combustible sources
- Reduction of the concentration of free radioactive material in the confinement boundary
- Protection against external events
- Radiation shielding

The BWRX-300 PCS:

- Encloses and supports the NBS, the RPV and connected piping systems.
- Provides radiation shielding.
- Serves as a boundary for radioactive contamination released from the NBS, or from portions of connected systems located inside the PCS.

##### **6.3.1.1 Energy Management**

Post-accident energy management is described in Subsection 6.3.3 for Passive Containment Cooling System (PCCS). The heat removal capabilities of the Passive Containment Cooling System (PCCS) have been analyzed and accepted by the US NRC in NEDC-33922P-A, Revision 3, *BWRX-300 Containment Evaluation Method* (Reference 6.3.2-1). The analysis of PCCS heat removal capabilities from normal operation, DBAs, or DECAs is addressed in Chapter 15, Section 15.5 for DSA. PCCS heat removal during severe accident (SA) or BDBA is confirmed in Chapter 15, Section 15.6 for Probabilistic Safety Assessment (PSA).

##### **6.3.1.2 Management of Radioactive Material**

For AOOs and DBAs, the PCS contains and confines any radioactive materials in accordance with REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.6 (Reference 6.3.2-2).

### **6.3.1.3 Management of Combustible Gases**

For DECAs and BDBAs, the control of combustible gases is analyzed in Chapter 15, Section 15.6 with a description of the combustible gas system in Subsection 6.3.5.

### **6.3.1.4 Management of Severe Accidents**

SA and BDBA management are analyzed and described in Chapter 15, Section 15.6 for PSA.

## **6.3.2 Primary Containment System**

PCS encompasses the RPV, attached piping, equipment, and the RIVs, making a leak-tight boundary. The PCS is the fourth fission product barrier (preceded by the fuel pellet, fuel clad, and RPV), and is also a floodable volume to assure core coverage in response to BDBAs. The PCS is a SC1 system that assures the integrity of the barriers prevents releases and maintains the plant in a safe state.

The leak-tight containment vessel confines fission product releases from the core and prevents the release of radioactive contamination to the environment. The PCS outlined in red on Figure 6.3.2-1 is a dry, inerted containment with active and passive cooling features that dissipates normal (power operation) and off-normal (transient or accident) heat loads. Table 6.3.2-1 provides key parameters used in the developing the BWRX-300 SCCV design.

### **6.3.2.1 Safety Design Bases**

The PCS meets the requirements of REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.6 (Reference 6.3.2-2):

- **Leak-Tight Boundary:** Provides a leak-tight boundary in conjunction with other safety class systems/features limiting the radiological effects of contamination released from the NBS and connected systems inside of containment to meet the requirements of REGDOC-2.5.2, Section 8.6.4 and Section 8.6.12. All boundary components including the SCCV inner wall, containment closure head, equipment and personnel access airlocks, electrical and piping penetrations ensure leak-tight integrity. The PCS establishes the maximum allowable leakage rate under specified parameters for all boundary forming components and provides containment pressure monitoring during all normal and off-normal conditions.
- **Structural Integrity:** Maintains structural integrity of the pressure boundary preventing structural failures. Structural pressure boundary components including the SCCV, containment closure head, equipment and personnel access airlocks, electrical and piping penetrations, and Containment Isolation Valves (CIVs) maintain structural integrity under all specified normal and off-normal static and dynamic loading conditions, to meet the requirements of REGDOC-2.5.2, Section 8.6.2 (refer to Chapter 9, Section 9B.2.1).
- **Capability for Pressure Tests:** the PCS is subject to pressure testing, at a specified pressure to demonstrate structural integrity. Testing is conducted before plant operation commences, and at appropriate intervals, to meet the requirements of REGDOC-2.5.2, Section 8.6.3 and 8.6.5. Refer to Subsection 6.3.7.
- **Radiation Shielding:** Provides permanent radiation shielding within and outside containment limiting doses ALARA. Containment shielding is provided within the RPV core region, and the SCCV structural concrete also functions as radiation shielding external to the containment, to meet the requirements of REGDOC-2.5.2, Section 8.13.

- Equipment and Personnel Access: Access openings are sized accommodating repair/replacement activities. A removable opening (containment closure head) in the upper containment region allows access to the RPV for refueling outage activities, to meet the requirements of REGDOC-2.5.2, Section 8.6.7.
- Structural Support: Provides structural support to safety class SSCs within containment, including the RPV, as well as anchorage for connected piping systems and components, to meet the requirements of REGDOC-2.5.2, Section 8.6.8.
- Off-Normal Containment Cooling: Maintains the pressure and temperature within acceptable limits through the PCCS during all off-normal conditions, to meet the requirements of REGDOC-2.5.2, Section 8.6.9.
- Leak Detection: Detects leaks within the containment boundary by means of pressure detection and containment sump level detection.

### **Steel-Plate Composite Containment Vessel Safety Design Basis**

The dry, inerted SCCV pressure containing single chamber configuration is designed to accomplish the following safety design basis for structural integrity:

- SCCV safety functions rely on passive design features and natural forces.
- SCCV structure design accommodates the full range of loading conditions consistent with normal plant operation, AOOs and all DBA design loads.
- SCCV structure is sized and equipped to contain the mass and energy released by a large-break LOCA, and for small breaks.
- SCCV structure accommodates the maximum external negative pressure difference relative to the enclosing structure surrounding the SCCV.
- SCCV is protected from or designed to withstand hypothetical missiles from internal and external sources or the uncontrolled motion of broken pipes.
- SCCV pressure boundary wall has an integral or external passive cooling, using air or water, necessary by design analysis and plant layout constraints.
- SCCV forms a fission products release barrier discharged from the RCPB into the SSCV volume as a result of an accident up to and including the defined DBA events.
- SCCV accommodates flooding for long-term core cooling and permits recovery actions up to and including fuel assembly removal following a postulated DBA.

The PCS is designed to accomplish its functions following the initiation of a DBA event. Refer to Table 6.3.2-1 for the preliminary primary containment key design parameters.

The containment performance is demonstrated in Chapter 15, Section 15.5 and 15.6. Refer to Chapter 15 for DEC, in accordance with REGDOC-2.5.2, Section 8.6.12.

#### **6.3.2.2 System and Equipment Descriptions**

The PCS is located in a Seismic Category A below-grade silo and constructed using Steel Bricks™ as a SCCV (Figure 6.3.2-1) within the Reactor Building (Refer to Chapter 3, Subsection 3.5.1). The PCS is part of the nuclear island design and arranged so that the major pools of the nuclear island are above containment. This layout provides protection for PCS and equipment and components from external hazards. This arrangement uses safety class passive cooling methods for the core, RPV and the SCCV.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

The PCS design uses a nitrogen-inerted containment atmosphere during most operating modes. Atmosphere control is provided by the Containment Inerting System (CIS) (refer to Chapter 9A, Section 9A.4.2 for the CIS system description). The CIS establishes and maintains an inerted nitrogen atmosphere for the prevention of hydrogen combustion in a post SA condition. The inert atmosphere design minimizes long-term corrosion and degradation of the SCCV and the contained components by limiting oxygen exposure during plant operating service life.

The PCS has provisions for personnel access and habitability during plant outages to perform maintenance, inspections, and tests required to assure SCCV integrity and reliability, and the integrity and performance reliability of interfacing SSCs contained inside the PCS boundary.

The PCS is comprised of the following components:

- Steel-Plate Composite Containment Vessel
- Containment Closure Head
- Personnel and Equipment Airlocks
- Mechanical Penetrations
- Electrical Penetrations
- Refueling Bellows Seal
- Integrated Leak Rate Penetration (refer to Subsection 6.3.7)
- Containment Corium Shield / Liner (refer to Chapter 15, Appendix 15A)
- Core Region Biological Shield (Bioshield, refer to Chapter 9B, Section 9B.2.2)
- Passive Containment Cooling System (refer to Subsection 6.3.3)
- Containment Support Structures
- Maintenance Structures
- RPV Vertical and Lateral Support Structures

Figure 6.3.2-1 and Figure 6.3.2-2 show a cross sectional views of containment within the reactor building, and Figure 6.3.2-3 is a simplified diagram of the containment SSCs.

The sub-compartments in containment are the compartments below the RPV and inside the RPV support structure, the spacing between the RPV and RPV Pedestal, and containment dome above the refueling bellows. Containment sub-compartments do not have large high energy pipes. Therefore, the sub-compartment boundaries are not subject to large transient loads resulting from pipe breaks inside the sub-compartments. Pipe breaks outside the sub-compartments may result in transient loads on the sub-compartment boundaries. However, since the pipe breaks occur in a large volume and since the sub-compartments have sufficiently large openings, these loads are not large in magnitude. The pressure differential across the boundaries is evaluated in Section 6.6.1 of NEDC-33922P-A (Reference 6.3.2-1) for the most limiting case and are small. The sub-compartment design includes the resulting differential pressures from pipe breaks outside the sub-compartments.

#### **6.3.2.2.1 Steel-Plate Composite Containment Vessel**

The SCCV is a concrete and steel structure that is a leak-tight containment boundary, providing structural support and radiation shielding external to containment.

The containment structure is composed of an SCCV, which consists of a Steel-Plate Composite cylindrical wall, basemat, top slab, and steel containment closure head. The containment structure is completely enclosed within the deeply embedded RB that includes containment penetrations and other safety components. The containment structure houses the Steel-Plate Composite internal pedestal that supports the RPV and also provides shielding. The SCCV is also integrated with the reactor building, and the integrated structure is supported by a common Steel-Plate Composite basemat.

The SCCV is classified as a SC1 and Seismic Category A structure. The structure description and analysis comply with all applicable codes and standards provided in Chapter 3, Subsections 3.5.3. The SCCV structure is discussed in Chapter 9B, Section 9B.2.1.

The SCCV is designed considering the rules and requirements of ASME BPVC, Section III, Division 2, SCCV Code Case (Reference 6.3.2-3).

#### **6.3.2.2.2    *Containment Closure Head***

The containment closure head is a removable steel dome that covers the opening in the top slab of the SCCV that is over the RPV functioning as a portion of the upper containment boundary. The interface between the closure head and the SCCV is a metal ring that is anchored into the top slab of the SCCV and includes a bolted flange for attachment/removal of the closure head.

The containment closure head is removed during reactor refueling and replaced using the RB crane prior to reactor operation.

The containment closure head forms part of the reactor cavity pool and retains water above PCS during normal operation. The containment closure head is SC1 and Seismic Category A structure.

#### **6.3.2.2.3    *Personnel and Equipment Airlocks***

Two containment airlocks function as a portion of the containment boundary when closed, one in the upper elevation and one in the lower elevation. The containment airlocks are sized to accommodate removal of the most limiting components.

#### **6.3.2.2.4    *Containment, RPV and Maintenance Support Structures***

The internal structural steel is SC1 and Seismic Category A structure that is described in Chapter 9B, Section 9B.2.2.

### **Containment Support Structures**

The internal structural steel, which includes the Containment Equipment and Piping Support Structures (CEPSS) and support floors at level -21 and -29 m, are steel structures located inside containment (refer to Figure 6.3.2-1). The containment internal steel structures are supported by the SCCV wall, RPV pedestal, and the CEPSS which also interface with the RPV at the upper stabilizer locations. The internal structural steel consists of various structural components such as beams and columns.

Refer to Chapter 9B, Section 9B.2.2 for a description of the structural role of the support floors at Level -21 m and -29m, and the primary CEPSS functions.

### **Maintenance Structures**

Platforms, stairs, ladders, or other structural facilities required for personnel entry and work areas performing inspections, tests, and maintenance or modifications on equipment within containment are provided.

Space is provided around equipment locations inside the SCCV allowing removal, servicing, and maintenance. Platforms and staircases for equipment access allow inspection and maintenance.

Platforms are provided for equipment inspection, examination, surveillance, and maintenance. Platforms and structures do not hinder the SCCV performance and considers the effects of high energy jet and impingement loads minimizing missile and debris generation. Removable stairs and platforms are used in place of permanent installations as needed.

The SCCV has installed crane rail(s) and cart track(s) (typically, monorail or mono-track) and pick points for lifting, positioning, and transporting components, equipment, maintenance tools, materials, inspection and test machines, equipment and tools, servicing systems and components inside the SCCV including the interior side of the SCCV boundary.

### **RPV Vertical and Lateral Support Structures**

The RPV pedestal is a SC1 and Seismic Category A structure.

The RPV pedestal is a cylindrical-shaped, steel-plate composite structure that structurally supports the RPV. The RPV pedestal, enclosed within the SCCV, is welded to a Steel-Plate Composite basemat that is connected with the SCCV.

Openings are provided in the RPV pedestal to permit pipe routing to the RPV, permit in-service inspection of the RPV and piping, and ensure personnel access into the under-vessel region.

The RPV Pedestal primary safety functions are:

- Provide structural support to SSCs such as the RPV, RPV stabilizers and miscellaneous platforms.
- In conjunction with the Bioshield, the RPV Pedestal provides radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOOs, DBAs, and DECAs.

Additional description of the RPV support is provided in Chapter 5, Section 5.9. The primary functions of the Bioshield are described in Chapter 9B, Section 9B.2.2.

#### **6.3.2.2.5 Refueling Bellows Seal**

The refueling bellows seal is designed as a mechanical appurtenance to the RPV and forms a seal between the outside RPV wall surface and the upper region of the SCCV during refueling operations when the RPV head is removed.

The refueling bellows assembly accommodates the movement of the vessel caused by operating temperature variations and seismic activity. The refueling bellows provides a 360° structural barrier that retains the refueling cavity water above the SCCV when the SCCV head is removed. The refueling bellows design includes protection from puncture or damage from dropped items during refueling outage activities, or workers performing RPV or SCCV head removal or installation activities. The design includes a drain to remove water from the bellows low point that can be cleaned (i.e., for removal of non-soluble radioactive contamination, including fuel particles, that settle onto the bellows assembly during refueling outages).

#### **6.3.2.2.6 Mechanical Penetrations**

Mechanical penetrations function as a portion of the containment boundary consisting of a metal sleeve that is welded into the SCCV and an attached section of process piping, allowing transfer of the process medium across the containment boundary when isolation valves are open. High temperature mechanical penetrations have an air gap between the sleeve and the process pipe. Cold mechanical penetrations have no air gap and the sleeve itself functions as the process pipe.



As indicated above, containment penetrations are outlined in the PCS Simplified Facility Diagram in Figure 6.3.2-3. Mechanical penetrations are designed maintaining containment integrity under DBAs conditions, including pressure, temperature, and radiation.

All mechanical penetrations are SC1 and Seismic Category A or B structures. Piping systems that penetrate containment pressure boundary are equipped with provisions to automatically and reliably seal preventing leakage.

The SCCV is periodically tested to measure the integrated leakage rate from the SCCV structure to confirm the leak-tight integrity of the pressure boundary.

Refer to Figure 6.3.2-3 for the PCS Simplified Facility Diagram that includes mechanical penetrations.

#### **6.3.2.2.7 *Electrical and Instrument Sensing Line Penetrations***

Electrical penetrations consist of a metal sleeve which is welded into the SCCV. The penetration assembly forms a leak-tight seal with the containment boundary, while allowing the passage of continuous electrical cables across the boundary.

Instrument fluid sensing line penetrations are arranged in ganged-penetration assemblies.

#### **6.3.2.3 Primary Containment System Materials**

Refer to the Subsection 6.3.2.2 for component materials. PCS materials are discussed in Chapter 9B, Section 9B.2.1 in accordance with REGDOC-2.5.2, Section 8.6.11 (Reference 6.3.2-2).

#### **6.3.2.4 Interfaces with Other Equipment or Systems**

Systems interface with the SCCV by directly communicating with the containment atmosphere or penetrating the SCCV pressure boundary for equipment connections located inside the SCCV. See Table 6.3.2-2 for PCS interfaces, Figure 6.3.2-4 and Chapter 9B, Section 9B.2.1.

#### **6.3.2.5 System and Equipment Operation**

The PCS is a passive system.

Containment is inerted with nitrogen supplied by the CIS (refer to Chapter 9, Section 9A.4.2) within 24 hours after the plant reaches 15% reactor power level, in accordance with the OLCs. Prior to reactor shut down, containment is supplied with breathable air during the de-inerting process up to 24 hours prior to descending to 15% reactor power level, in accordance with OLCs. During normal plant operation, CIS maintains the SCCV slightly above atmospheric pressure conditions preventing non-inert air infiltration. Normal containment cooling is performed by the Containment Cooling System (CCS) (Chapter 9A, Section 9A.5.6).

PCS leakage is monitored using the following parameters:

- Containment sump pump-out rate
- Containment sump level changes
- Air-handling unit (AHU) drainage flow rates
- Containment atmosphere fission product sampling

The PCS pressure and bulk average and individual sensor temperature readings are continuously monitored with alerts for pressure and temperature above normal range.

The reactor cavity above the containment is used as a heat sink for long-term cooling in abnormal or accident conditions if the CCS fan coolers become unavailable. The PCCS removes heat from the containment and reduces the containment temperature and pressure in abnormal and accident conditions.

Heat transfer occurs from containment to the PCCS by natural convection and condensation through the PCCS and containment dome to the subcooled water in the equipment pool. There are no active components or actuation signals required, as the PCCS is always in-service.

Fan coolers provide the primary means of heat removal from containment during normal operation (Chapter 9A, Section 9A.5.6 for CCS). Heat is lost to the reactor cavity pool through the containment head.

#### **6.3.2.6 Instrumentation and Control**

Instrumentation for interfacing systems is described in Table 6.3.2-2. Refer to Chapter 12, Subsection 12.3.14 for the Containment Monitoring Subsystem (CMon).

#### **6.3.2.7 Monitoring, Inspection, Testing, and Maintenance**

Containment monitoring, inspecting, testing and maintenance is described in Subsection 6.3.7.

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets.

#### **6.3.2.8 Radiological Aspects**

The DSA for the PCS and SCCV are described and analyzed in Chapter 15, Section 15.5.

#### **6.3.2.9 Performance and Safety Evaluation**

Containment design pressure and temperature are established with sufficient margin to the calculated peak pressure and temperature for large-break LOCA events. Refer to Chapter 15, Subsection 15.5.4 for large-break LOCA.

The peak containment pressure and temperature mass and energy release resulting from a large-break LOCA are limited by fast closing RIVs.

The PCS total leakage following a DBA is limited to less than leakage rates that result in offsite doses greater than those set forth by REGDOC-2.5.2, Section 4.2.1 (Reference 6.3.2-2). Refer to Chapter 15, Section 15.3 for safety acceptance criteria and Chapter 15, Section 15.7 for the results of both the DSA and PSA compliance with the acceptance criteria.

In the event of pipe failure, the PCS does not have sub-compartments containing large, high-energy pipes preventing dynamic loads from sub-compartments pressurization. The containment shell and the internal structures withstand jet loads resulting from large pipe breaks. The sub-compartments have large openings to the rest of containment preventing a large differential pressure from developing. The pressure differential across the sub-compartment boundaries is evaluated in Section 6.6.1 of NEDC-33922P-A for the most limiting case and found to be small (Reference 6.3.2-8). Refer to Chapter 15, Section 15.5 for further analysis.

**6.3.2.10 References**

- 6.3.2-1 NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC.
- 6.3.2-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 6.3.2-3 ASME BPVC-III, "BPVC Section III – Rules for Construction of Nuclear Facility Components, Division 2, "Code for Concrete Containments," American Society of Mechanical Engineers.

**Table 6.3.2-1: Preliminary Primary Containment Key Design Parameters**

Parameter	Value and Units <sup>5</sup>
Free Containment Volume (estimated)	7685.3 m <sup>3</sup> (271,404 ft <sup>3</sup> )
Minimum Free Volume (design)	6960 m <sup>3</sup> (245,790 ft <sup>3</sup> )
Containment Height, excluding the dome	37.985 m (124.6 ft)
Containment Diameter (outer)	17.30 m (56.8 ft)
Containment Shell Plate Thickness	15 mm (0.59 in)
Containment Single Brick Thickness	610 mm (24.02 in)
Containment Closure Head Diameter	7.84 m (25.72 ft)
Number of Manholes on Refueling Bellows	2 (Assumed)
Diameter of the Manholes	0.762 m (2.5 ft) (Assumed)
Containment Atmosphere	Nitrogen with less than 4% Oxygen
Normal Pressure	1.72 - 9.0 kPaG (0.25 - 1.30 psig)
Containment Design Internal Pressure <sup>(1)</sup>	413.7 kPaG (60.0 psig)
Max. Bulk Average Temperature <sup>(2)</sup> (Normal Operating Temperature)	57.2 °C (135 °F)
Max. Average Temperature – Upper Containment	65 °C (150 °F)
Max. Average Temperature – Lower Containment	43 °C (110 °F)
Peak Transient Shell Temperature: 0.0 – 0.01 h	195 °C (383 °F)
Design Temperature: 0.01 - 72 h <sup>(3)</sup> (Design Temperature)	166 °C (330 °F)
Containment Atmospheric Moisture <sup>(4)</sup>	10% Relative Humidity (RH) – 100% RH

- (1) Design pressure is conservatively established with approximately 10-15% margin above the initial thermal-hydraulic analysis peak pressure and the historical BWR Mark I and II containment design pressure values.
- (2) Bulk temperature represents normal plant operating conditions based on historical BWR steel containment temperature records. Bulk average temperature is not indicative of local component thermal environment maximum operating temperature that is evaluated separately.
- (3) Peak transient temperature is used for environmental qualification. Peak temperature for the most severe conditions of BDBAs may exceed this value for a short duration.
- (4) The SCCV atmospheric RH is a low for normal plant operations without leaks. The range is used for component environmental qualification. 100% RH is assumed for corrosion evaluation. Humidity is not specifically controlled during any plant operating mode, but in the dry containment design it is a monitoring parameter for unidentified leakage.
- (5) Numeric values presented are developmental and unconfirmed.

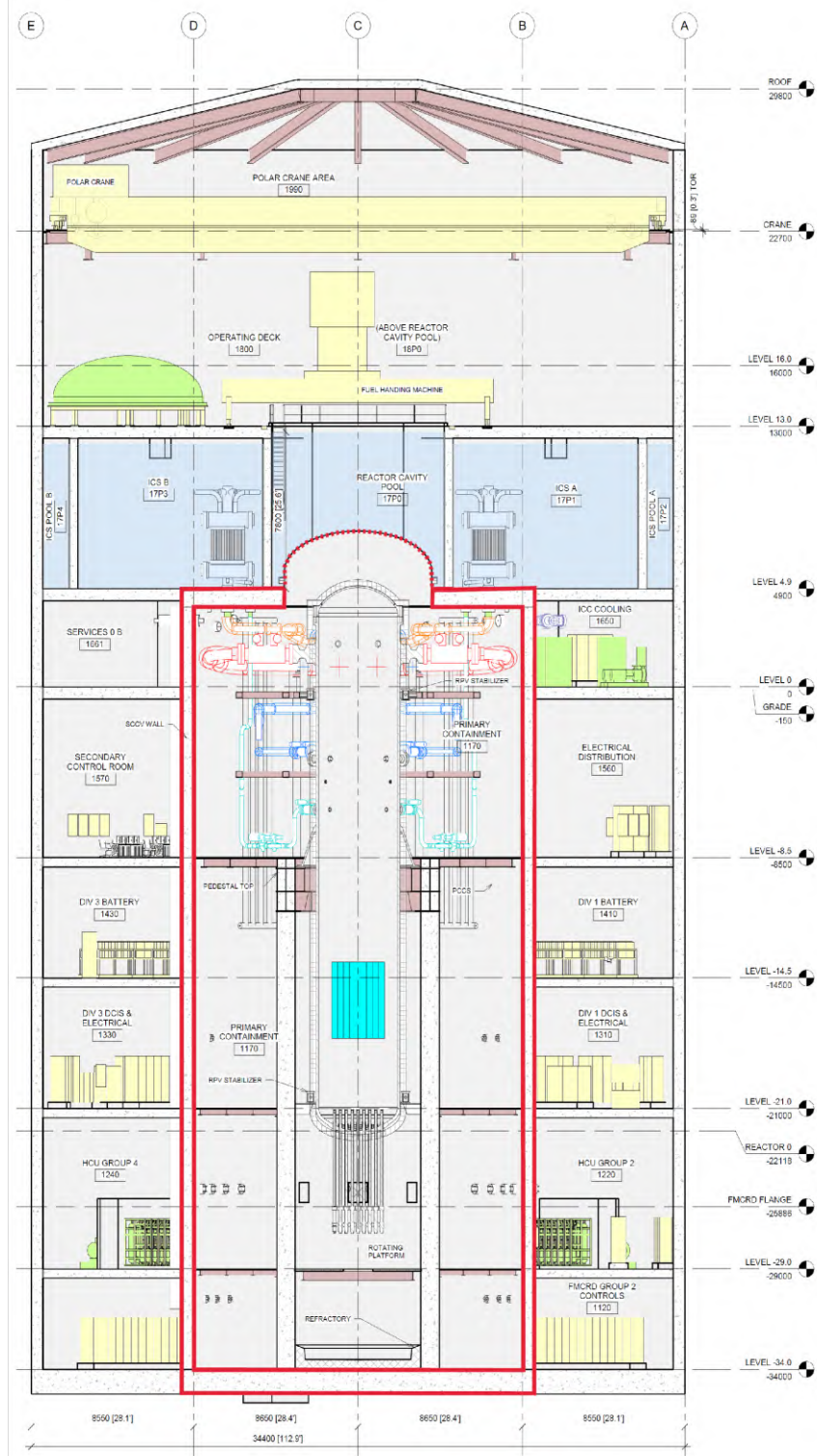
**Table 6.3.2-2: Primary Containment System Interfaces**

Interfacing System	Interface Description
Multiple Systems	Process systems penetrating containment provide containment isolation. Reference Penetration Interfaces on Figure 6.4.2-2: Primary Containment System Simplified Diagram
Multiple Systems	PCS provides structural support for process piping systems within containment
Nuclear Boiler System	NBS provides attachment / support of refueling bellows seal
Nuclear Boiler System	PCS provides vertical and lateral RPV support
Safety Class 1 Instrumentation and Control System	SC1 Instrumentation and Control System receives / provides signals to SC1 instrumentation and controls in PCS
Non-Safety Class (SCN) Instrumentation and Control System	SCN Instrumentation and Control System receives / provides signals to SCN instrumentation and controls in PCS
Process Radiation and Environmental Monitoring System (PREMS)	PREMS monitors containment atmosphere for oxygen, hydrogen, temperature, pressure, humidity, water level, area radiation, and gross gamma radiation
Fuel Pool Cooling and Cleanup System	Fuel Pool Cooling and Cleanup System provides supply water to PCCS
Fuel Pool Cooling and Cleanup System	PCCS returns heated water into the equipment pool

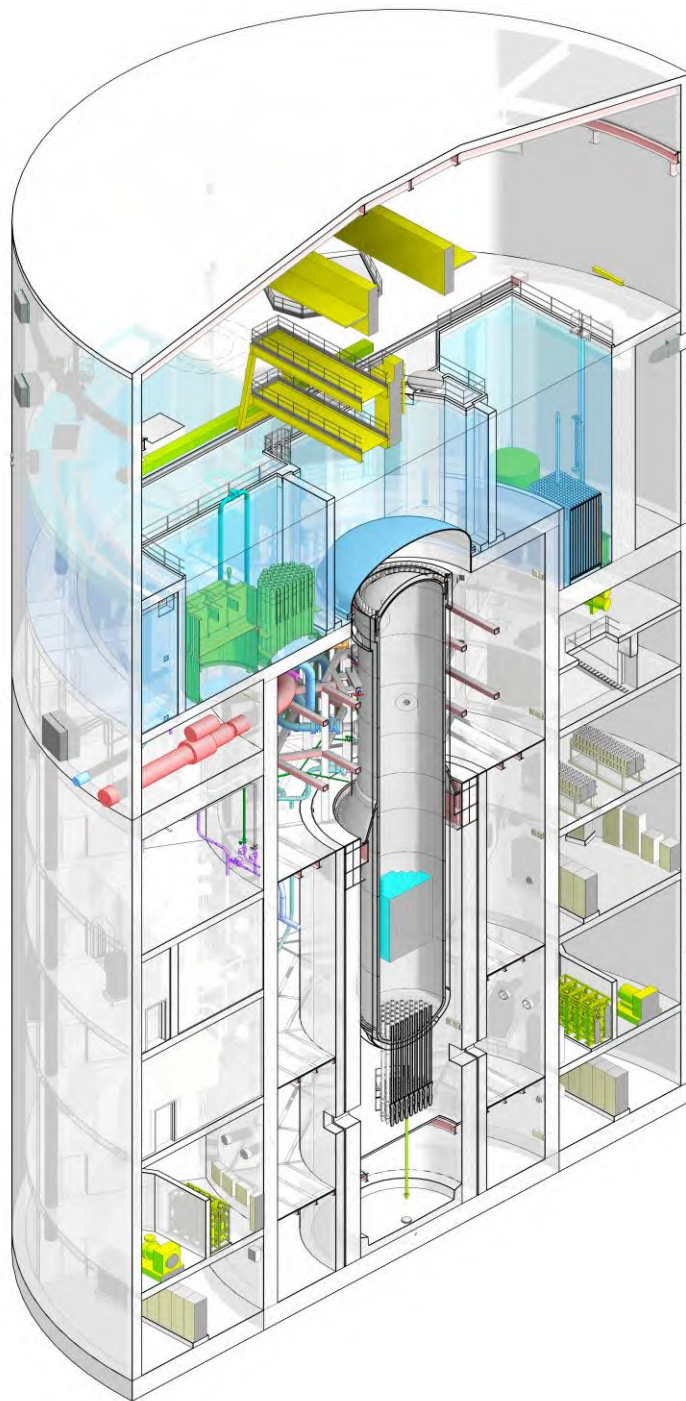
NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description
Non-Safety Electrical Distribution System	SCN Electrical Distribution System provides power to SCN electrical loads in PCS
Containment Inerting System	CIS supplies nitrogen or breathable air to containment during inerting and de-inerting evolutions
Containment Inerting System	CIS exhausts nitrogen or breathable air from containment during inerting and de-inerting evolutions
Containment Inerting System	CIS provides Containment Overpressure Protection
Containment Cooling System	CCS maintains containment temperature within normal operating range
Equipment and Floor Drain System	PCS monitors outflow from containment sump and sump level to allow trending and identification of excessive containment leakage
Equipment and Floor Drain System	PCS provides containment sump structural support
Reactor Building Structure	RB provides structural SCCV support

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 6.3.2-1: General Containment Arrangement – Reactor Building Section View**



**Figure 6.3.2-2: General Arrangement – Reactor Building Section View**



NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

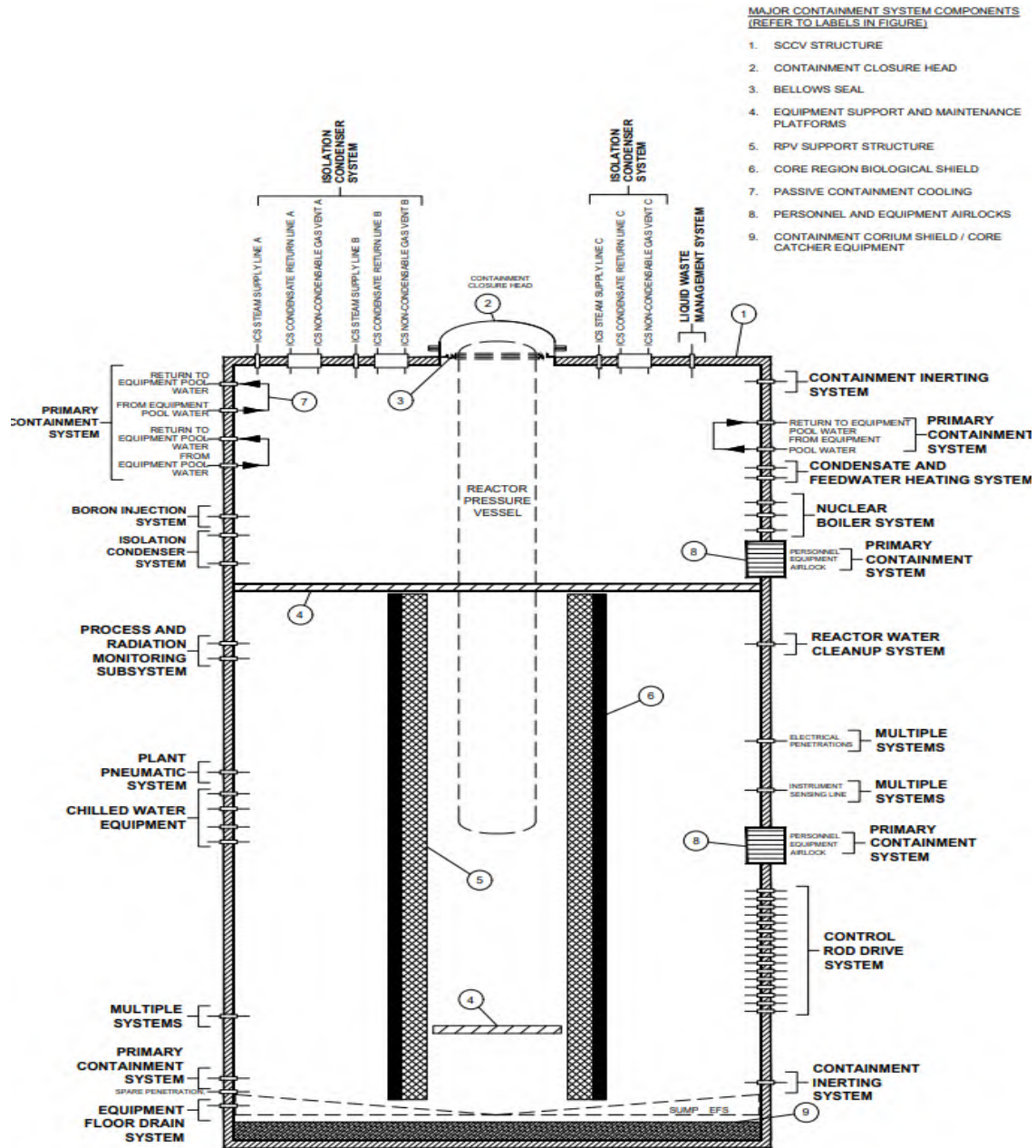


Figure 6.3.2-3: Primary Containment System Simplified Diagram  
(Containment Mechanical Penetrations)

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

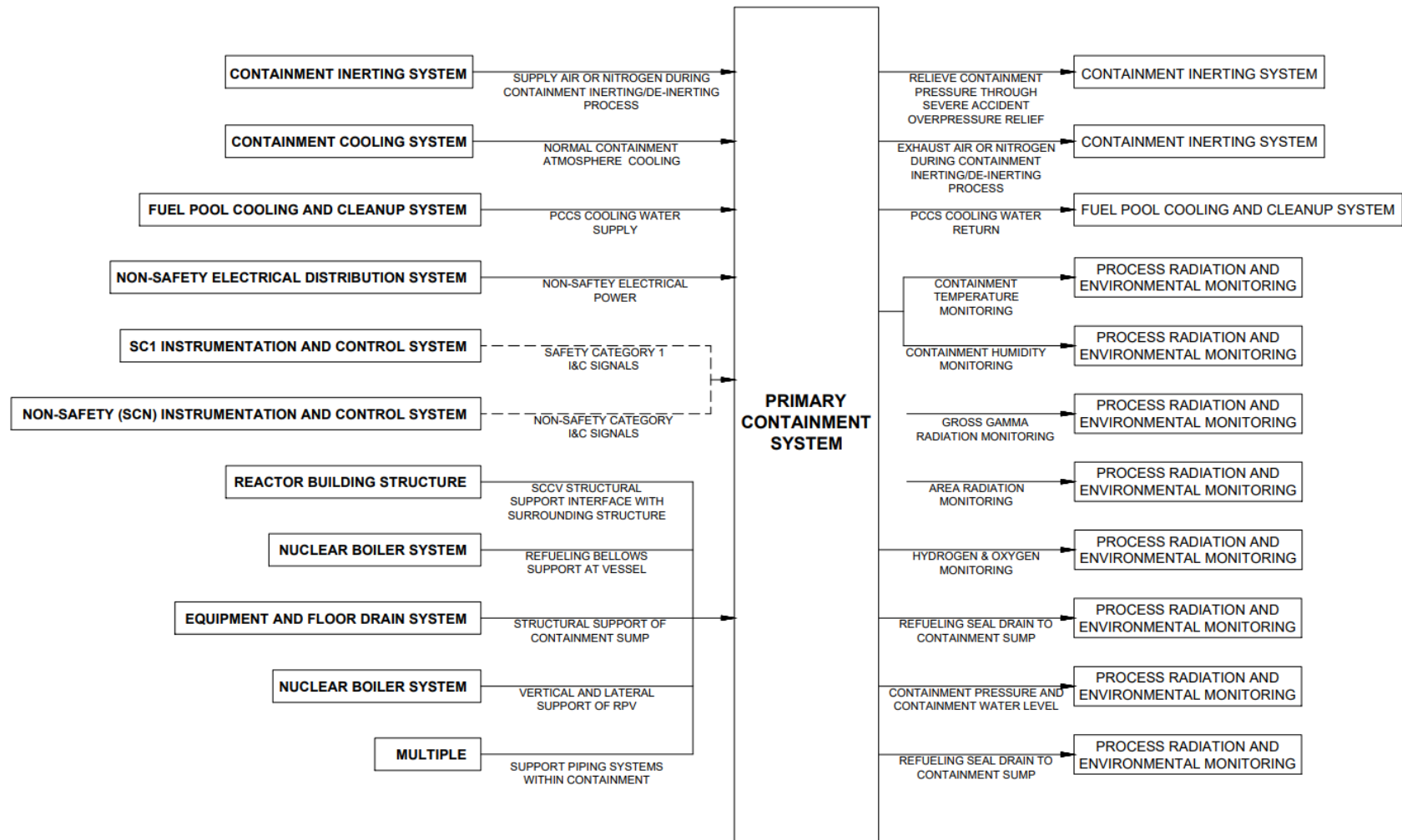


Figure 6.3.2-4: Primary Containment System Interfaces

### **6.3.3 Passive Containment Cooling System**

The PCCS maintains temperature and pressure of the PCS through passive cooling during off-normal conditions. The PCCS is based upon proven concepts and simple thermosyphon principles. Heat is rejected to the equipment pool above containment by natural circulation using three independent trains of PCCS pipes as depicted in Figure 6.3.3-1. In Hot Shutdown, Startup, and Power Operation, the reactor cavity equipment pool gate is removed, connecting the equipment pool and reactor cavity pool to provide a heat sink for PCCS. There are no active components or actuation signals required, as the PCCS is always in-service. During normal operation, PCCS does not contribute to heat removal significantly since containment temperature is maintained by the CCS (Chapter 9A, Section 9A.5.6). The PCCS becomes effective when steam is discharged into the containment following a DBA pipe break.

#### **6.3.3.1 Safety Design Bases**

The PCCS removes heat from the containment and reduces the containment temperature and pressure in accident conditions. The reactor cavity and equipment pools above the containment are used as a heat sink for long-term cooling following DBAs or if the CCS fan coolers become unavailable.

#### **6.3.3.2 Description**

The PCCS consists of three independent trains of eight connected pipes with a top and bottom header connected to the equipment pool, which communicates with the reactor cavity pool when the PCCS is required to be operable.

The PCCS units are placed at locations that are not subject to jet loads or protected against them.

Each PCCS train has an isolation valve on the supply and return header outside containment that can be used to manually isolate a train in the event of a PCCS pipe leak or to remove a train from service for maintenance or testing during Cold Shutdown or Refueling Modes when the system is not required. The valves are normally open and are remotely operated from the Main Control Room (MCR). The PCCS isolation valves are located outside containment as close to the containment boundary as practical.

The PCCS inlet piping, on each train features a drain valve attached to the piping upstream of the isolation valve. The drain valve enables each PCCS train to be drained to the Liquid Waste Management System for maintenance and testing during the Refueling Mode when the equipment pool is drained.

The PCCS piping outside the containment boundary is open to the equipment pool. The ends of the supply and return pipes in the equipment pool are flanged / threaded with filters ensuring the piping remains free of debris. Each PCCS train can be pressure-tested during the Refueling Mode after the equipment pool and PCCS system is drained.

#### **6.3.3.3 Materials**

The materials of PCCS are ASTM A312 / ASME SA312, GR TP316/ TP316L Seamless Stainless Steel (SS) nominal size piping.

#### **6.3.3.4 Interfaces with Other Equipment or Systems**

The PCCS is part of the PCS. PCCS interfaces with the equipment pool to maintain containment pressure and temperature within the design limits during accident conditions or loss of active containment cooling. The Fuel Pool Cooling and Cleanup System (FCP) as outlined in Table 6.3.2-2 provides water to the equipment pool and heat sink for PCCS. Refer to Figure 6.3.3-1 for the PCCS piping layout.

#### **6.3.3.5 System and Equipment Operation**

Heat transfer occurs from the containment to the PCCS by natural convection and condensation through the PCCS to the subcooled water in the equipment pool.

The PCCS becomes effective when steam is discharged into the containment following a pipe break. The steam discharged to containment raises containment temperature and increases the steam content for condensation to occur. Condensation heat transfer, even in the presence of non-condensable gas, is much larger than heat transfer from a non-condensable gas at the same temperature. Heat transferred to the PCCS from the containment is removed by the natural circulation of water in single phase flow and rejected to the subcooled water in the reactor cavity equipment pool.

Because the ICs have sufficient capacity to remove decay heat, the majority of the energy during an accident is removed directly from the RPV. PCCS must remove only the energy released from the break to the containment. The energy released from a break is limited for a large break since large breaks are isolated rapidly. Energy released from unisolated small breaks is also limited due to the small break size. Additionally, the ICs depressurize the RPV making the break flow very small within a few hours. Depressurization of the RPV also reduces the RPV temperature, reducing the heat load from hot surfaces. The PCCS has sufficient capacity to reduce the pressure and temperature in the containment, minimizing leakage following an accident and maintaining pressure and temperature below the design limits in isolation events. Calculations for large and small breaks are presented in Chapter 15, Subsection 15.5.4.

During Power, Hot Shutdown, and Stable Shutdown Operations, the PCCS is aligned to allow flow of the equipment pool water that is in communication with the reactor cavity pool through the closed loop within containment. The PCCS is not required to be in-service for Cold Shutdown Operation or in Refueling Mode.

#### **6.3.3.6 Instrumentation and Control**

There are no I&C inputs for PCCS as a passive system.

#### **6.3.3.7 Monitoring, Inspection, Testing and Maintenance**

Monitoring, inspection, testing, and maintenance for PCCS piping are planned and performed to meet system reliability and availability targets. The PCCS pressure test is performed during the modes when PCCS is not required to be operable.

The PCCS is an integral part of the containment boundary and is designed to be periodically pressure tested as part of the overall Containment Leakage Rate Test (Section 6.3.7) program to demonstrate the structural and leak-tight integrity.

#### **6.3.3.8 Radiological Aspects**

The DSA for AOOs and DBAs is discussed in Chapter 15, Section 15.5. No design basis AOOs or DBAs lead to core damage. Breaks outside containment are the only DBAs leading to a normal coolant concentration release evaluated in Chapter 15, Subsection 15.5.9, and this release would not affect the PCCS heat removal performance.

#### **6.3.3.9 Performance and Safety Evaluation**

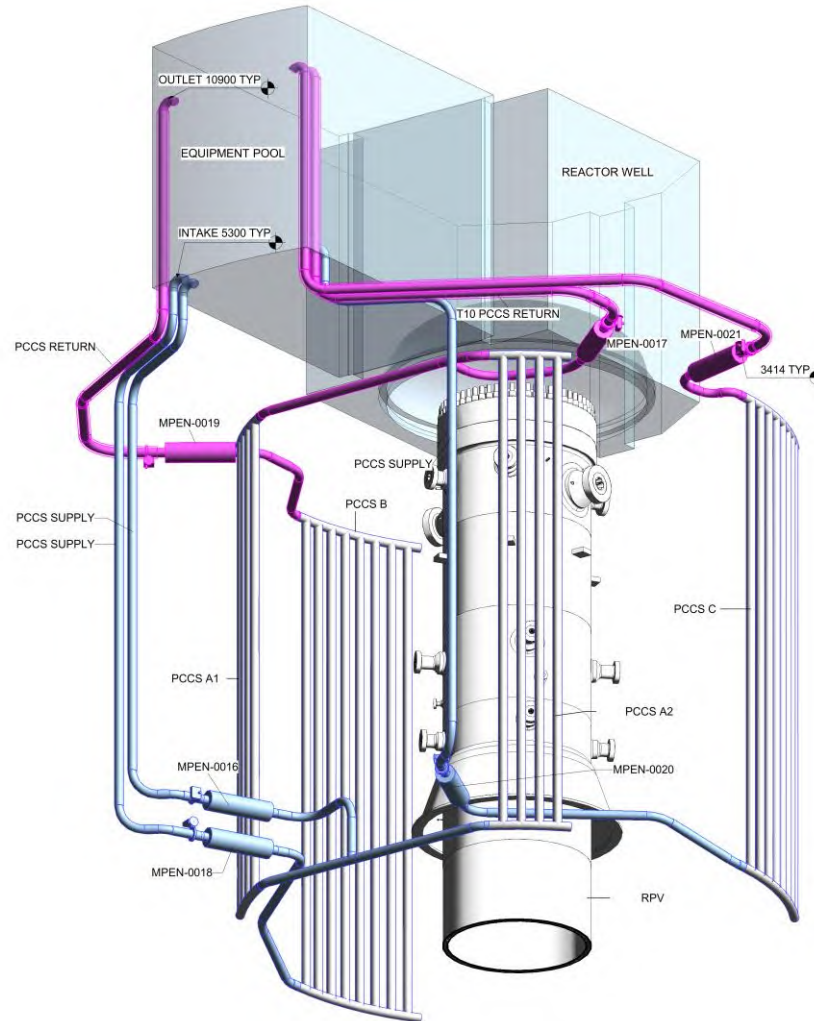
The PCCS transfers heat from the containment to the reactor cavity and equipment pools to maintain containment pressure and temperature within the design limits during accident conditions or during loss of active containment cooling. The PCCS functions for containment depressurization and heat removal are passive and do not require onsite or offsite electric power system operation.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

The limiting DBAs for containment pressure and temperature are large-break LOCAs from main steam or feedwater pipe breaks. The PCCS rapidly reduces containment peak pressure following a large-break LOCA. Small unisolated breaks may become more limiting in the longer term for containment structure temperature. Containment pressure and temperature resulting from AOOs are bounded by those resulting from DBAs. The PCCS removes the energy from containment. Refer to Chapter 15, Subsection 15.5.4 for further details on DBA LOCA analysis.

The PCCS does not have a high duty function for long-term containment cooling as most of the decay heat from the RPV is removed by the ICS. For RPV isolation and loss of all AC events, containment pressure and temperature are limited by condensation on containment walls and containment heat removal by the PCCS, and by RPV decay heat removal by the ICS. The PCCS rejects heat to the reactor cavity and equipment pools above containment during DBAs as outlined in Chapter 15, Section 15.5 (Reference 6.2-2). Further analysis on the performance of the PCCS is to be developed in detailed design and the results are incorporated in the POSAR.

GOTHIC is used to evaluate the BWRX-300 containment response to mass and energy release from the RPV. The performance of the PCCS is included in the overall containment evaluation method.



**Figure 6.3.3-1: Passive Containment Cooling System**

#### **6.3.4 Containment Isolation**

This section describes the isolation of mechanical systems penetrating primary containment. Containment Isolation Valves (CIVs) are SC valves that prevent the uncontrolled release of containment contents in the event of an accident or other conditions by maintaining the integrity of the containment boundary. Piping systems penetrating primary containment are provided with detection, isolation, and containment functions that are redundant, reliable, and proven performance. CIVs are periodically tested to validate operability and determine if valve leakage is within acceptable limits.

There are several systems with RIV inside containment. The safety analysis assumes that large pipe breaks are quickly isolated by the RIVs following a LOCA preserving reactor coolant inventory before there is any potential for a significant radiological release. For piping that have RIVs, the closest piping terminal end (high stress and fatigue location) to the RPV assembly is located outboard of each set of two in-series RIVs.

Automatic CIVs outside containment are also included even though probability of a pipe break inside containment is small. These CIVs outside containment area are also included for the lines with RIVs. Automatic CIVs outside containment are not fast closing because there is no credible scenario in which fission products greater than what is contained in normal reactor coolant is released to containment. The outside containment automatic CIVs closure time assures containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant in source term evaluations that are completed in the detailed design phase. These closure times are in the order of minutes. Additionally, the valve closing time for all CIVs supports specific break isolation functions balanced with water hammer and valve loading considerations. Small pipes for level instruments use Excess Flow Check Valves (EFCVs).

##### **6.3.4.1 System and Equipment Functions**

The containment isolation function maintains containment integrity by providing protection against the uncontrolled release of radioactive materials from the containment to the environment as the result of an accident.

Lines penetrating containment that provide a potential path for unfiltered radioactive release are automatically isolated by initiating CIVs closure limiting the release of fission products during and after a postulated accident. Redundant CIVs are included in each system line near the containment boundary that close on predefined parameters to prevent releases from containment, and any negative impacts from the system in that line. Penetrations are included at the boundary between the line and the containment limiting leakage at the connection. Penetrations are designed to maintain structural integrity during the extreme environmental conditions resulting from a postulated accident and protect against leakage. The CIVs, piping between the CIVs and penetrations, and the penetrations are included as a part of the containment boundary. together supporting the limited release of fission product leakage during and after an accident.

##### **6.3.4.2 Safety Design Bases**

All mechanical systems with lines penetrating the containment boundary have a SC function supporting containment integrity by isolating and limiting the release of radioactive leakage to within allowable limits during and following an accident. CIVs are periodically tested confirming operability and determining that valve leakage is within acceptable regulatory limits. Containment penetrations with thermal sleeves have resilient seals and expansion bellows permitting periodic inspection and testing.

The containment isolation design complies with the following:

- CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.6.6 on Containment Isolation and 8.6.10 (Reference 6.3.4-1)
- CNSC REGDOC-1.1.2, Licence to Construct a Reactor Facility, Section 4.5.9 Safety system and safety support systems (Reference 6.3.4-2)

CIVs in lines penetrating the containment boundary meet Canadian regulations and guidance for containment isolation providing protection against uncontrolled releases to the environment and isolating systems that may negatively affect containment. The containment isolation provisions for the ICS steam supply and condensate return lines constitute an appropriate application of the alternative means of meeting REGDOC-2.5.2, Section 8.6.6 (Reference 6.3.4-1). A single failure does not disable the containment isolation function. The two in-series RIVs that function as CIVs remain open during accident conditions allowing the ICS to function as ECCS.

CIV classification is defined by Safety Class, Seismic Category, Quality Group, and location as described in Chapter 3, Section 3.2. Refer to Chapter 15, Section 15.2 for the BWRX-300 Fault Evaluation process. The CIV DL functions are described in the individual fault sequence events in Chapter 15, Section 15.5.

Containment integrity is ensured by containment penetrations, RIVs, CIVs, and the containment piping up to CIVs are designed to Seismic Category A and B per REGDOC-2.5.2 (Reference 6.3.4-1). The penetrations, piping up to CIV's, and valves are designed to ASME BPVC-III NB, Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB - Class 1 Components (Reference 6.3.4-3).

#### **6.3.4.3 Containment Isolation Valves**

The following mechanical systems penetrate the containment boundary that require containment isolation:

- Nuclear Boiler System (NBS)
- Process Radiation and Environmental Monitoring System (PREMS)
- Isolation Condenser System (ICS)
- Boron Injection System (BIS)
- Control Rod Drive System (CRD)
- Reactor Water Cleanup System (CUW)
- Condensate and Feedwater Heating System (CFS)
- Chilled Water Equipment (CWE)
- Plant Pneumatic System (PPS)
- Containment Inerting System (CIS)
- Equipment and Floor Drain System (EFS)
- Water, Gas, and Chemical Pads (WGC)
- Passive Containment Cooling System (PCCS)



## **Mechanical System Interface CIVs**

Mechanical system interface for CIVs meet the requirements of the International Atomic Energy Agency (IAEA) SSG-53, Section 4.154 (a) and Section 4.154 (b), where applicable (Reference 6.3.4-4).

### *Nuclear Boiler System*

Two MSLs and a RPV head vent that are included in the RCPB are part of the NBS that penetrates containment. The MSLs CIVs fail in the closed position, with valve actuators that maintain the valves closed by positive mechanical means. This containment isolation is single-failure proof. The main steam CIVs are fast closing and fail-closed type valves.

### *Process Radiation and Environmental Monitoring System*

The Continuous Hydrogen and Oxygen Sample Panel Train A and B lines penetrate containment and are directly connected to the containment atmosphere.

### *Isolation Condenser System*

The ICS steam supply and condensate return lines penetrate containment and are a part of the RCPB. Each ICS train (one steam supply lines and one condensate return line per train) has two inboard automatic isolation valves.

Vent lines from the ICS tie into MSLs between inboard and outboard CIVs and do not penetrate the containment. Each ICS train vents to the MSL and has two series inboard automatic isolation valves.

Each ICS train provides an interface to the RPV with either BIS or SDC. The interface piping is routed through a containment penetration and includes two in series outboard remote actuated isolation valves. The actual ICS to SDC boundary occurs at the outlet of the most outboard isolation valve for each train (refer to Section 6.2 for ICS).

### *Boron Injection System*

A boron injection line has a remote air-operated outboard CIV and a check valve as the inboard containment and RIV. The injection location is in the ICS "C" loop condensate return line downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location into the ICS condensate return line requires a separate containment penetration for the BIS injection line (refer to Chapter 15, Appendix 15B Complementary Design Features for BDBAs). The air-operated injection valve, when open, provides a flow path to the reactor for injecting a boron solution or demineralized water into the reactor. The injection valve, when closed after an injection event, allows re-establishing containment isolation. Locating a BIS CIV inside containment when needed for beyond design basis accident mitigation is not practical. The ICS RIVs are required to be open for accident mitigation.

### *Control Rod Drive System*

Hydraulic lines for the FMCRD scram function use penetrations without isolation valves based on the closed system piping outside primary containment and RCPB isolation uses internal ball check valves in the drive design. The CRD system and the associated hydraulic insertion line performs a safety critical function by providing high-pressure water to produce a reactor scram. Each FMCRD includes an integral ball check valve at the drive flange insert port. The check ball operation serves to plug the insert line port and limits the reactor coolant discharged in the event of an insert line break. Additionally, manual isolation valves may be used to further isolate the Hydraulic Control Unit (HCU) from the hydraulic insertion line if needed.

### *Reactor Water Cleanup System*

The CUW mid-vessel suction lines penetrate the containment and are a part of the RCPB. Each CUW mid-vessel suction line has two inboard automatic isolation valves. Both CUW mid-vessel suction lines combine inside containment and have one outboard automatic isolation valve for the combined header. The CUW containment isolation valves close upon receiving an isolation signal from SC1, SC2 and SC3 control platforms. The CUW CIVs fail in the closed position, with valve actuators maintaining the valves closed by positive mechanical means.

### *Condensate and Feedwater Heating System*

The FW line CIVs fail in the closed position, with valve actuators maintaining the valves closed by positive mechanical means. Each CFS line has two automatic isolation valves inside containment and two automatic isolation valves outside containment.

### *Chilled Water Equipment*

The CWE supply and return lines penetrate containment and are directly connected to the containment atmosphere if equipment leakage occurs. Each CWE supply line and return line has two inboard and one outboard automatic isolation valve.

### *Plant Pneumatic System*

Containment PPS includes two distinct sections: one supplies nitrogen and instrumentation air connections to the pneumatic valves inside containment and the other supplies service and breathing air inside containment during plant outages. The nitrogen and service air section are aligned to containment during normal operations. The breathing air section is used during plant outages and isolated during normal operations. Both sections are directly connected to the containment atmosphere. Isolation valve arrangements for these two distinct sections:

- Each CIV for the pneumatic nitrogen and instrumentation air section line has at least one automatic CIV inside and outside containment.
- Each CIV for service and breathing air section lines have one locked closed isolation valve inside containment and one automatic isolation valve outside containment. A simple check valve is not used as the automatic isolation valve outside containment.

### *Containment Inerting System*

Each CIS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. Both isolation valves on these lines are located outside the containment vessel to remove them from the harsh containment environment and protect them from the effects of flood and dynamic effects of pipe breaks.

The valves are located as close as practical to the containment vessel. The piping from the containment vessel up to and including both valves is an extension of the primary containment boundary and is designed in accordance with ASME BPVC-III NB-2021 requirements (Reference 6.3.4-3). The isolation valves arrangement and connecting piping allows for a single active failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, that does not prevent CIS containment isolation.

#### *Equipment and Floor Drain System*

The EFS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. There are two CIVs placed in series, located outside containment that are placed as close to the primary containment wall as practical. It is impractical to have an isolation valve inside containment above the floor elevation. The pressurized containment sump tank is credited for containment leak detection. The isolation valves on the EFS containment drain line are normally open to connect the drain line to the pressurized sump. The EFS CIVs close automatically. Upon loss of actuating power, the automatic isolation valves fail closed, providing a greater safety position.

#### *Water, Gas, and Chemical Pads*

The WGC demineralized water supply line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. CIVs include two manual isolation valves with one inboard and one outboard of containment. The WGC System demineralized water manual CIVs are locked closed, only opened during Modes 5 or 6 and are administratively controlled

#### *Passive Containment Cooling System*

Each PCCS (Section 6.3.3) train has an isolation valve on the supply and return header outside containment to isolate a train in the event of a PCCS pipe leak or remove a train from service for maintenance or testing during Cold Shutdown or Refueling Modes when the system is not required. The isolation valves are normally open and are remotely operated from the MCR. The PCCS isolation valves are located outside containment and as close to the containment boundary as practical. Supply and discharge connections from the pool are connected to closed-loop piping within containment.

### **6.3.4.4 Mechanical Penetrations**

Penetrations provide the connection between the containment and penetrating lines to prevent the uncontrolled release of containment contents in the event of an accident or other conditions inside the containment. Refer to Figure 6.3.4-1 for the penetrations view without containment wall.

### **6.3.4.5 Instrument Line Penetrations**

Instrument lines penetrating containment are sized, or an orifice installed that ensures that following a line break outside primary containment during normal operation the leakage is minimized. The rate and extent of coolant loss is within the normal reactor coolant makeup capabilities and the integrity and functionality of containment air treatment systems is maintained. Excess flow check valves are commonly used in BWR fleet instrument lines that:

- Do not close accidentally during normal operation
- Will close readily if the instrument line integrity outside containment is lost during normal operation or accident conditions
- Re-open or can be re-opened when conditions allow

Instrument line penetrations are either single or multiple. The multiple instrument line penetrations have 4, 6, or 8 instrument lines per penetration. Wherever possible, instrument line penetrations are shared minimizing the total quantity of penetrations used. Different systems can share the same penetration but only one system is declared for the penetration designation.

Containment isolation function is applied to all mechanical instrument sensing line penetrations of the SCCV boundary providing the highest reliability for maintaining instrument function while limiting potential radioactive releases if an instrument line is ruptured outside the SCCV boundary.

#### **6.3.4.6 Penetration Sleeves**

Penetration sleeves are used for high energy lines to reduce the effects of high temperature and/or pressure of the penetrating pipe on the containment concrete.

High energy lines are defined for normal plant conditions in system operation where either condition is met:

- Maximum operating temperature exceeds 95°C.
- Maximum operating pressure exceeds 1.9 MPaG. This definition does not apply to accident conditions.

For the BWRX-300 design, each mechanical system line penetration is categorized as either a hot penetration or cold penetration.

Cold penetrations are directly embedded into the SCCV. Hot penetrations are provided with a thermal sleeve that attaches to the SCCV to minimize the conductive heat transfer to the SCCV wall. Penetration sleeves are also used for the CWE lines even though they are not considered high energy lines.

Mechanical system penetrations containing trapped liquid between the CIVs have features that relieve thermally-induced pressurization that comply with IAEA SSG-53, Section 4.160 (Reference 6.3.4-4) that states: "Overpressure protection should be provided for closed systems that penetrate the containment and for isolated parts of piping that might be over pressurized by an increase of the temperature inside the containment atmosphere in accident conditions." A self-relieving penetration is typically selected, and the inboard isolation valve is oriented so that excess fluid is released inward to the containment. Using a separate relief valve provides penetration piping overpressure protection is permissible on a case-by-case basis when no other isolation valve selection option is available.

Mechanical system penetrations arrangements provide clearance for inspections.

#### **6.3.4.7 Containment Isolation Support Piping**

The mechanical system piping supporting containment isolation functions for those systems listed in Subsection 6.3.4.3 is the piping between the inboard and outboard CIVs for most system applications.

#### **6.3.4.8 Materials**

Containment integrity is ensured by the SCCV penetrations, piping, and valves design to ASME BPVC-III NB-2021 – Class 1 Components (Reference 6.3.4-3) with Seismic Category B for valves and Seismic Category A for penetrations and piping (Reference 6.3.4-1).

#### **6.3.4.9 Interfaces with Other Equipment or Systems**

Interfacing mechanical systems penetrating the containment boundary that require isolation are discussed in Subsection 6.3.4.3.

#### **6.3.4.10 System and Equipment Operation**

System and equipment operation is discussed in the following Subsection 6.3.4.11.

#### **6.3.4.11 Instrumentation and Control**

Containment isolation occurs automatically from closure signals generated by analytical limits or by remote manual. CIV closure is completed once an isolation signal is received. After an isolation valve closes, the valve does not open until the signal is removed, or the operator takes action by resetting the switch. The reactor operator cannot override a containment isolation signal to return the valve to the normal operating position by a single action.

##### ***6.3.4.11.1 Isolation Signals***

Isolation valves that are closed automatically for mechanical system lines that penetrate containment are capable of being reopened in the presence of an isolation signal. The controls for resetting an isolation signal do not result in the automatic reopening of CIVs.

Position indication for all power operated CIVs are provided in the control rooms. Position indication for power operated CIVs are based on actual valve position and not on demanded valve position. CIVs are configured to permit visual verification of valve position. Check valve CIVs are equipped with closed and open position indication devices, unless justified as not applicable.

#### **Instrument Lines**

Instrument lines penetrating the primary containment are provided with an automatically operated CIV, or one that an operator can manually operate from a remote location, or an excess-flow check valve. Self-actuated excess flow check valves in instrument lines are designed to close when the flowrate increases to a value representative of a loss of piping integrity outside containment.

Self-actuated excess flow check valves in instrument lines have the capability to automatically reopen when the pressure in the instrument line is reduced after previously closing due to an increase flowrate that was representative of a loss of piping integrity outside containment.

##### ***6.3.4.11.2 Containment Isolation Valves Closure Times***

For system lines that have an open path from the containment to the environment, closure times are defined based upon the deterministic safety analysis (Chapter 15, Section 15.5) minimizing the radiological effects in the event of an accident. The outside containment automatic CIV closure times assures containment isolation prior to the first fission product release that is greater than what is contained in normal reactor coolant source term evaluations. For some cases, the closure times are also determined based on minimizing the effects of containment flooding due to a pipe break within containment.

Total closure times consider both the I&C total time to initiate the valve actuation, and the mechanical total time for the valve to full close:

- I&C time – time it takes for the instrument to identify the trip signal to the actual switch that trips the valve actuator
- Mechanical time – time it takes the actuators to move the valve to fully closed

The following general criteria are used in establishing closure times:

- Established closure times prevent allowable radiological releases from being exceeded.
- For large breaks, the RIVs close rapidly and prevent significant loss of RPV inventory, uncovering the reactor core, or fuel clad heat-up. The largest diameter steam line and FW pipes are selected for DSA are the most limiting RPV blowdown conditions for plant isolation response.

#### **6.3.4.11.3 Valve Position**

For the following plant operational modes, CIVs have a typical valve position:

- Normal Operation
- Shutdown
- Post-Accident
- Loss of Motive Power

The positions are typical and based on plant operation and may vary based on system mode changes.

On loss of motive power each CIV takes the position that provides greater safety.

#### **6.3.4.11.4 Position Monitoring**

The containment isolation function, including provisions for control, indication, and performance under loss/restoration of power conditions, is instrumental in maintaining containment barrier integrity and not interfering with flow paths essential for reactor core cooling.

Position indication for all power-operated CIVs are provided in the MCR. Position indication for power-operated CIVs are based on actual valve position and not on demanded valve position. CIVs are configured to permit visual verification of valve position.

Manually operated valves are locked closed and are administratively controlled.

#### **6.3.4.12 Allowable Leakage**

The BWRX-300 CIVs minimize leakage. This section defines requirements specific to leakage performance. Allowable leakage for containment leak rate testing is provided before pre-operational testing. A discussion on containment leakage testing is included in Subsection 6.3.7.

##### **System Specific Allowable Leakage Requirements**

Leak rates are determined by manufacturer accepted seat leakage test methods for each systems specific design parameters.

##### **General Allowable Leakage Requirements**

Test requirements are determined for preoperational and periodic verification of primary containment SSCs to confirm leak-tight integrity and establishing test acceptance criteria. These tests are performed to assure that (a) leakage through the primary reactor containment and systems and components does not exceed allowable leakage rates values and (b) periodic reactor containment penetrations and isolation valve surveillance is performed so that any necessary maintenance and repairs are identified.

#### **6.3.4.13 Monitoring, Inspection, Testing and Maintenance**

All CIVs have individual leakage tests that are performed supporting verification that containment leakage is within the allowable limits. All CIVs are leak tested as discussed in Section 6.3.7, containment leakage testing in accordance with REGDOC-2.5.2. Design features required to support testing are defined within this section.

To support containment leakage testing, CIVs are manually cycled in the same manner that the valve closes upon receipt of an isolation signal. For example, if a valve closes automatically by pneumatic operation from an isolation signal, then the valve is also closed remotely by manual pneumatic operation.

Mechanical pipelines penetrating containment contain test connections that support CIV leakage testing. This test is performed by applying a test pressure through one test connection in the same direction the valve performs its safety function. A second test connection is located on the opposite side of the valve for line venting.

#### **6.3.4.14 Radiological Aspects**

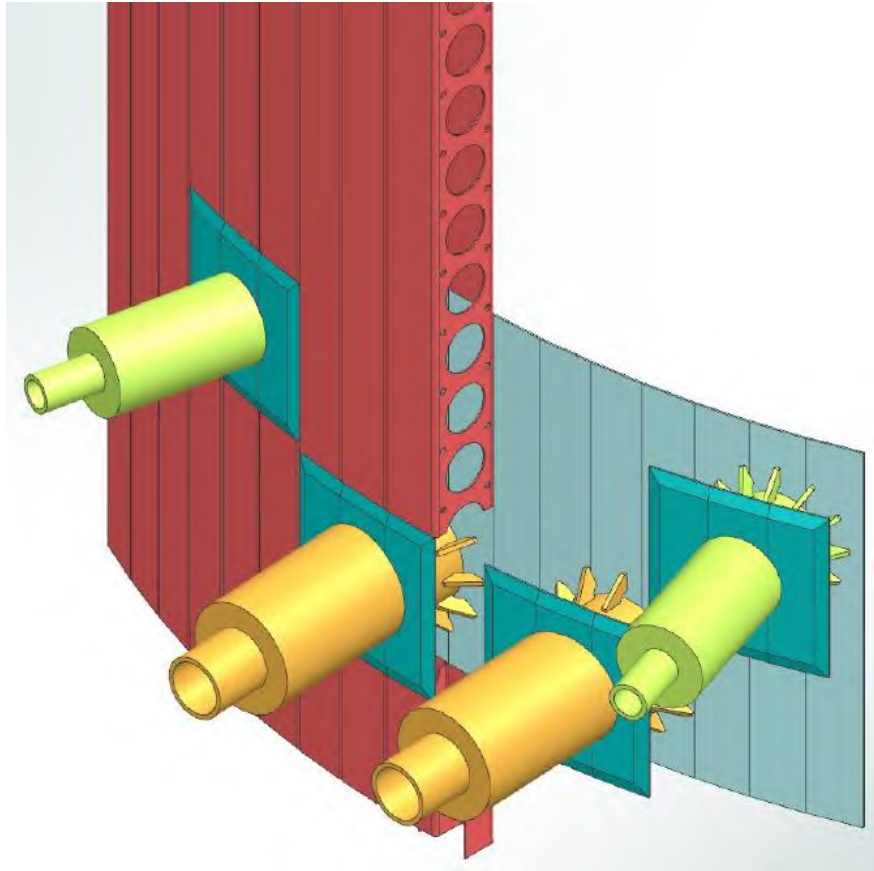
Penetrations are arranged to provide sufficient space and shielding ensuring planned maintenance and operations is performed minimizing personnel exposure.

#### **6.3.4.15 Performance and Safety Evaluation**

The performance requirements (time to closure) for CIVs are confirmed by the deterministic safety analysis in Chapter 15, Section 15.5.

#### **6.3.4.16 References**

- 6.3.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 6.3.4-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence to Construct a Reactor Facility,"
- 6.3.4-3 ASMR BPVC Section III, "Rules for Construction of Nuclear Power Plant Components," Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 6.3.4-4 , IAEA Safety Standards Series No. SSG-53, "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants," International Atomic Energy Agency.



**Figure 6.3.4-1: Penetration View Without Containment Wall**



### **6.3.5 Systems for Protection Against Overpressure and Underpressure**

The CIS provides overpressure protection for BDBA. This complementary design feature is discussed in Chapter 15, Appendix 15B Complementary Design Features for Beyond Design Basis Accidents. The containment overpressure vent flow path is a hardened vent in SA cases where containment failure from overpressure may occur.

### **6.3.6 Filtered Venting System**

Refer to Chapter 15, Appendix 15B for the filtered venting system, which is to be further analyzed to determine if filtered venting is required for BDBA.

### **6.3.7 Containment Leakage Testing**

Containment Leakage Rate Testing is performed to assure leakage through containment and systems and components penetrating the SCCV does not exceed allowable leakage rate values as specified in plant OLCs and associated bases. The testing measures the rate at which a contained air mass escapes through the containment boundary at a specific pressure using sensitive instrumentation as described in 10 CFR 50, Appendix J (Reference 6.3.7-1) and ANSI/ANS-56.8 (Reference 6.3.7-2). Additionally, periodic surveillance of SCCV penetrations and isolation valves is performed to verify that proper maintenance and repairs are made during the service life of the SCCV.

The Integrated Leak Rate Test (ILRT) consists of the containment leakage rate and local leakage rate testing (LLRT). Testing requirements are in accordance with 10 CFR 50, Appendix J (Reference 6.3.7-1), and ANSI/ANS 56.8, Section 3.2.3 (Reference 6.3.7-2).

This testing is required on all boundaries that serve as barriers to the release of primary containment atmosphere following a design-basis LOCA as identified in 10 CFR 50 Appendix J III.D.2(b)(i) and 10 CFR 50 Appendix J III.C.1 (Reference 6.3.7-1).

Testing is performed prior to placing containment into service and periodically during plant operation, as part of the in-service testing examination and testing program ANSI/ANS-56.8-2020, Section 3.2.2 (Reference 6.3.7-2), and 10 CFR 50 Appendix J III.D.1 (Reference 6.3.7-1), respectively.

#### **6.3.7.1 Preoperational Leakage Rate Testing**

The preoperational ILRT determines whether SCCV structures comply with specified strength and design requirements in accordance with the requirements of ANSI/ANS-56.8, Section 3.2.2 (Reference 6.3.7-2). The preoperational proof and leakage rate testing is performed upon completion of construction and prior to criticality in accordance with the requirements of ANSI/ANS-56.8, Section 3.2.2 (Reference 6.3.7-2).

The preoperational ILRT includes general visual inspection of the accessible interior and exterior surfaces of the SCCV and components prior pressurization. The visual inspection is performed to identify deterioration that may affect leak tightness in accordance with ANSI/ANS-56.8-2020, Section 3.2.1 (Reference 6.3.7-2).

#### **6.3.7.2 Periodic ILRT**

Periodic ILRTs is performed by scheduled test frequency after the SCCV is placed into service. The test includes visual inspection of SCCV as described in Subsection 6.3.7. The test includes a positive pressure test of the containment and penetrations. Results of visual examination and ILRT are reported according to the requirements in ANSI/ANS-56.8-2020, Section 5.11.1 (Reference 6.3.7-2).

### 6.3.7.3 Periodic ILRT Requirements

Containment integrity is verified through pre-operational and periodic leak rate testing using established acceptance criteria as identified in ANSI/ANS-56.8 (Reference 6.3.7-2) and USNRC 10, Appendix J (Reference 6.3.7-1). Testing requirements includes:

- Leakage test requirements
- Test instrumentation
- Test procedures
- Test methods
- Acceptance criteria
- Data analysis
- Inspection and recording test results
- Guidance on required component and pathway testing
- Test frequency

LLRT confirms the SCCV leak-tight integrity boundary in accordance with:

- CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.6.4 (Reference 6.3.7-3)
- USNRC 10 CFR 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors* (Reference 6.3.7-2)
- ANSI/ANS-56.8, *Containment System Leakage Testing Requirements* (Reference 6.3.7-1)

ILRT procedures are written to satisfy the test method guidance of ANSI/ANS-56.8, (Reference 6.3.7-2) and is consistent with USNRC 10, Appendix J (Reference 6.3.7-1). The procedures are used to measure leakage rate of the SCCV boundary and evaluate the results to ensure the maximum allowable containment leakage is not exceeded. Refer to Chapter 3, Section 3.10 for a discussion on in-service monitoring, tests, maintenance, and inspections.

### 6.3.7.4 ILRT Examination Methods

The ILRT test method is the absolute method of leakage determination using the mass plot analysis technique for containment ILRT in accordance with ANSI/ANS-56.8, Section 5.6.2 (Reference 6.3.7-2). The tests are to confirm that the actual containment leak rate does not exceed the maximum allowable leakage at pressure used in the plant safety analysis to demonstrate that release limits are not exceeded, as per ANSI/ANS-56.8, Section 6.4 (Reference 6.3.7-2).

Results of the containment ILRT are validated by the performance of a verification test as per the guidance ANSI/ANS-56.8, Section 5.9.4 (Reference 6.3.7-2). This may be achieved by one of the following methods:

- A. A superimposed known leakage rate
- B. The use of two independent measurement systems to monitor and collect leakage rate data

Verification of test acceptance criteria is performed in accordance with ANSI/ANS-56.8, Section 5.9.4 (Reference 6.3.7-2).

Following a controlled pressure change, the pressure and temperature is allowed to stabilize in accordance with ANSI/ANS-56.8, Section 5.9.3 (Reference 6.3.7-2).

#### **6.3.7.5 Duration of ILRT**

ILRT duration is determined by the ILRT lead based on the allowable leakage rate and the accuracy of measurements established for each test procedure in accordance with ANSI/ANS-56.8, Appendices (Reference 6.3.7-2).

#### **6.3.7.6 Isolation, Repair, or Adjustment to Leakage Path**

Any modification component replacement or repairs or adjustment which could affect SCCV integrity requires appropriate testing to demonstrate that the affected components meet the applicable leakage requirements ANSI/ANS-56.8, Section 2 (Reference 6.3.7-2). These include:

- Any requirements for closure of all containment valves, doors, and hatches
- Requirements for any test boundary isolation that differs from normal operation

#### **6.3.7.7 Pressurized Components**

The primary containment is pressurized at a safe rate with air that is clean, relatively dry, and free of contaminants in accordance with ANSI/ANS-56.8, Section 5.5 (Reference 6.3.7-2).

Sources of compressed gas are isolated or removed otherwise a compressed gas leakage rate test phase is conducted to ensure that valid leakage rates are being measured in accordance with ANSI/ANS-56.8, Section 3.2.6 (Reference 6.3.7-2).

#### **6.3.7.8 Containment Isolation Valve Closure**

Closure of penetration isolation valves is accomplished by normal means and without any preliminary exercising or adjustment. Exercising valves for the purpose of improving leakage performance is not permitted. Repairs of malfunctioning or leaking valves shall be made as necessary as per the guidance in ANSI/ANS-56.8, Section 3.3.3 (Reference 6.3.7-2).

#### **6.3.7.9 Acceptance Criteria**

Acceptance criteria ensures that the measured containment boundary leakage rate does not exceed maximum allowable containment leakage at test pressure. This limit demonstrates that release limits are not exceeded, in accordance with ANSI/ANS-56.8, Section 5.8 (Reference 6.3.7-2).

#### **6.3.7.10 Reporting of Periodic ILRT**

Reports documenting the results of each pre-operational and periodic tests includes a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable. Additionally, for the periodic test, the report shall include an analysis and interpretation of the ILRT and a summary analysis of LLRTs that were performed since the last ILRT in accordance with 10 CFR 50 Appendix J V.B.1 and 10 CFR 50 Appendix J V.B.2 (Reference 6.3.7-1).

**6.3.7.11 References**

- 6.3.7-1 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."
- 6.3.7-2 ANSI/ANS 56.8, "Containment System Leakage Testing Requirements," American Nuclear Society.
- 6.3.7-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

## 6.4 Control Room Habitability

The BWRX-300 design does not consider the MCR habitability a SC1 function. The Secondary Control Room (SCR) is physically and electrically separate from the MCR as required by REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*. However, in accordance with REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, Section 8.10.2, a physically and electrically separate Secondary Control Room (SCR) is provided in the RB when the MCR becomes uninhabitable. Events that necessitate evacuation to the SCR include fire, malevolent acts, and aircraft impact, as example.

The MCR located in the Control Building (CB) and the SCR located in the RB provide environments to protect the operators during all operational states and maintain the reactor in a safe condition. No DBA and DEC can simultaneously affect both control rooms to the extent that the FSFs cannot be performed as required by REGDOC-2.5.2, Section 8.10 (Reference 6.4-1).

Control Room Habitability (CRH) refers to the conditions required for life support and safe, effective operation of the plant during operational states and accident conditions. CRH is served by a combination of individual systems that collectively provide habitability. These systems are listed below:

- Heating Ventilation and Cooling System (HVS)
- Process Radiation and Environmental Monitoring System
- Lighting System
- Fire Protection System (FPS)

Habitability features include missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, and fire protection.

The design bases and descriptions of the various habitability features are contained in the following sections:

- Conformance with Applicable Regulations, Codes and Standards Section 1.11
- Evaluation of Site-Specific Hazards Section 2.2
- Protection Against External Hazards Section 3.3
- Protection Against Internal Hazards Section 3.4
- Control Room Habitability Function Section 6.4
- Control Building Heating, Ventilation and Air-Conditioning System Subsection 9A.5.2
- Fire Protection System Section 9A.6
- Lighting and Emergency Lighting Systems Subsection 9A.9.2
- Process Radiation Monitoring Subsystem (PRM) Section 11.5
- Design for Radiation Protection Section 12.3
- Shielding Section 12.4
- Area Radiation Monitoring Subsystem (ARM) Subsection 12.3.14

The MCR and SCR instrumentation is described in Chapter 7, Section 7.5, and Section 7.6, respectively. The CB structural evaluation for external hazards is provided in Chapter 3, Section 3.3 Protection Against External Hazards. Refer to Chapter 9B, Section 9B.3.2 for the CB structural integrity, which houses the MCR. There are no fission product releases resulting from a DBA or AOO as described in Chapter 15, Section 15.5.

In addition to the MCR and SCR, the BWRX-300 design incorporates emergency support facilities as required by REGDOC-2.5.2, Section 8.10.3 (Reference 6.4-1) and described in Chapter 19 (referred to as emergency response facilities in REGDOC-2.10.1). Habitability design requirements for emergency support facilities are included in POSAR in support of the Licence to Operate application.

#### **6.4.1 System and Equipment Functions**

The fundamental design philosophy is that the plant is operated from the MCR and only evacuated if necessary. For most PIEs, the operators remain in the MCR to safely operate the plant. If the MCR becomes uninhabitable, is expected to become uninhabitable, or functionality is unacceptably impaired, plant control is shifted from the MCR to the SCR. SCR habitability ensures the plant is maintained in a safe shutdown condition. The facility cannot be operated at power from the SCR.

A qualified route exists from the MCR to the SCR for events which necessitate evacuation of the MCR. An alternate route exists for fire events. MCR and SCR habitability provisions are provided to ensure that continued occupancy in one of the two locations is possible under PIEs for a minimum of 72 hours as required by REGDOC-2.5.2 (Reference 6.4-1).

Refer to Chapter 2, Section 2.2 for the evaluation of site-specific hazards which includes toxic gas assessments. The MCR is the assured operating location for smoke/external fire and toxic gas release event.

Both the MCR and SCR are protected for internal and external hazards and radiological events. The control room design features are based on proven technologies and human factors engineering considerations that are described in Chapter 18, Section 18.3.

A description of equipment to support CRH in the MCR and SCR is detailed below.

##### **6.4.1.1 Main Control Room**

As indicated above, the MCR, located in the CB, is the primary location for plant operators.

The CB is normally air conditioned and heated by 2x100% capacity chilled water supplied AHUs located on the CB roof, discharging through supply ductwork and a return plenum.

To accommodate off-normal MCR habitability contingencies, the following air handling units are provided for the CB:

- CB Toxic Gas Filtration Units (TGFUs) operate automatically when toxic gas is detected at the CB AHU outside air intakes. In this event, normal outside air supply to the operating CB AHU isolates and the TGFU discharge damper opens, allowing the associated TGFU to supply pressurization air to the CB through the normal CB supply AHU that continues to operate during a toxic gas event.
- Main Control Room Envelope (CRE) Emergency Filter Units (EFU) operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE isolation mode initiates when normal CB supply and return air dampers close. The operating CB normal supply AHU intake de-energizes upon high radiation detection and auto-start of the standby unit is defeated. Battery Room exhaust fans continue to operate based on timers.

#### **6.4.1.2 Secondary Control Room**

The SCR, located in the RB, is the assured shutdown location for the plant if habitability and control from the MCR is lost.

The SCR performs the following functions when the MCR is inhabitable:

- Initiate shutdown of the reactor and maintain the plant in a safe shutdown condition
- Monitor FSFs

#### **Emergency Filtration Units**

The SCR is provided with 2x100% EFUs and pressurization fans that supply ventilation air to the operators when automatically placed into service upon radiation, or smoke detection in the normal supply duct. A loss of power to both normal supply AHUs initiates operation of the SCR EFUs and pressurization fans. These two pressurization fans draw outside air through dedicated ducting with blast resistant openings.

#### **Normal Ventilation**

The SCR is normally provided filtered, conditioned air from the operating RB Lower-Level AHU. Normal Ventilation is separate from the emergency ventilation and ensures that the SCR temperature is maintained in a specified range. Electric duct heaters assist in maintaining SCR temperature in a normal range.

#### **Isolation Dampers**

The SCR makeup air handler intake ducts have isolation dampers that are closed on high radiation, smoke, or toxic gas to protect personnel from these hazards.

#### **6.4.2 Safety Design Bases**

Habitability provisions ensure that continued occupancy in one of the two locations remains possible under PIE as required by REGDOC-2.5.2, Section 8.10.2 (Reference 6.4-1). The design provides provisions for both internal and external events that pose a direct threat to continued operation of the MCR and SCR, and practicable measures to minimize the effects of AOOs, DBAs, or DECAs.

MCR and SCR habitability provisions ensure that continued occupancy in one of the two locations is possible under PIEs for a minimum of 72 hours as required by REGDOC-2.5.2 (Reference 6.4-1).

The following Canadian regulations, standards and guidance are evaluated and applied to the BWRX-300 MCR and SCR design:

- CNSC REGDOC-1.1.2, *Licence to Construct a Reactor Facility*, Section 6.15 Control room facilities (Reference 6.4-2)
- CNSC REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants* (Reference 6.4-1)
  - Section 7.15 Civil structure
  - Section 8.10.1 Main control room
  - Section 8.10.2 Secondary control room
- CNSC REGDOC-2.4.1, *Deterministic Safety Analysis* (Reference 6.4-3)
  - Section 4.3.2 Anticipated operational occurrences and design-basis accidents
- CSA N293, Fire Protection for Nuclear Power Plants and N293S1:21, Supplement No. 1 to N293-12 (Reference 6.4-4 and 6.4-5)

#### **6.4.2.1 Main Control Room and Secondary Control Room Design Basis**

The BWRX-300 design provides an MCR, from which the plant can be safely operated as required by REGDOC-2.5.2, Section 8.10.1 (Reference 6.4-1). The design provides an SCR, from which the plant can be placed and kept in a safe-shutdown state, when habitability is lost in the MCR, as required by REGDOC-2.5.2, Section 8.10.2 (Reference 6.4-1):

- Radiation exposure in the MCR, and SCR when applicable, to personnel throughout the duration of the postulated DBA does not exceed the dose acceptance criteria of CNSC REGDOC-2.4.1, Section 4.3.2 (Reference 6.4-3).
- The MCR and SCR habitability design features detect and protect personnel from smoke and airborne radioactivity. The MCR habitability design features additionally detect and protects personnel from toxic gas.
- The MCR and SCR habitability requirements are satisfied without the need for individual breathing apparatus or special protective clothing.

Because the BWRX-300 design does not consider control room habitability a SC1 function, the EFUs are not SC1 and are therefore not credited for mitigation of DBAs.

#### **6.4.3 Control Room Descriptions**

The habitability aspects of MCR and SCR HVAC is discussed in Subsection 6.4.3. Descriptions of the control room HVS, FPS and Lighting System are found in Chapter 9A, Sections 9A.5.2, 9A.6 and 9A.9.2, respectively. Descriptions of the ARM and PRM are found in Chapter 11, Section 11.5, and Chapter 12, Subsection 12.3.14. Refer to Chapter 9A, Figure 9A.5.1-1 for the RB HVAC Process Flow Diagram.

##### **6.4.3.1 Main Control Room Description**

The layout of the MCR habitability area in the CB is shown on Figure 6.4-1 to Figure 6.4-3.

CRE isolation and CRE EFU operation provides a pressurized envelope relative to adjacent spaces, maintaining CRE habitability during an airborne radiation release event.

The Main CRE is the area maintained for habitability purposes. Figure 6.4-1 depicts the Main CRE.



### **Emergency Filtration Units**

Main CRE EFUs operate automatically upon high radiation level detection at the operating CB supply AHU outside air intake. The CRE initiates isolation mode when normal CRE supply and return air dampers close. Automatic operation of the CRE EFUs provide suitable atmosphere for habitability and maintains the CRE slightly pressurized. The AHU de-energizes upon high radiation detection at the intake and auto-start of the standby unit is defeated.

### **Radiation Protection**

The CRE isolation and EFU operation provides a pressurized envelope relative to adjacent spaces, maintaining CRE habitability during a radiological release.

### **Fire Protection and Toxic Gas**

The normal outside air intakes are monitored for toxic gases and smoke and isolates the outside air dampers if toxic gas or smoke is detected.

#### **6.4.3.2 Secondary Control Room Description**

The SCR, located in the RB (refer to Figure 6.3.2-1) remains habitable for at least 72 hours following an event requiring activation of the SCR.

The layout of the SCR is shown in Figure 6.4-4.

### **SCR Emergency Filtration Units**

The EFUs outside air supply portion of the control room HVS is SC1 and Seismic Category B. Single active failure protection is provided by using two trains that are physically and electrically redundant and separated. In the event of one train failure, the failed train is automatically isolated, and the alternate train is automatically initiated. Each 100% capacity train is capable of supplying filtered air to the SCR pressure boundary at the required flow rate.

A small amount of leakage out through the SCR pressure boundary is anticipated through door seals, electrical penetrations, and isolation dampers. The exhaust from the SCR is optimized to ensure proper scavenging of air from the SCR in an amount equal to the supply. Backflow prevention through the controlled leak path is not required because the SCR is at positive pressure during normal and emergency operation.

The SCR EFU utilizes a High Efficiency Particulate Air (HEPA) filter, carbon filter, and HEPA post filter to provide radiological protection of the SCR outside air supply. The units, along with the associated intakes, dampers, and ductwork are SC1, Seismic Category B, and meet the requirements of CSA N293-12 (Reference 6.4-4). Safety class equipment such as the SCR EFUs are qualified to operate in the harsh environment that will be encountered at specific locations following a DBA or DEC. Each of the two SCR EFU trains incorporate a 100% capacity pressurization fan powered by separate division power supply. The SCR EFU trains ensure adequate fresh air is delivered and mixed in the SCR habitability area.

### **SCR Pressurization Fans**

The SCR is provided with EFUs and pressurization fans that supply ventilation air to the operators when automatically placed into service upon detection of radiation or smoke at the normal lower-level supply AHUs. A loss of power to both normal supply AHUs will also initiate operation of the SCR EFUs and pressurization fans.

## **Isolation Dampers**

The SCR habitability area pressure boundary includes penetrations, dampers, or valves, interconnecting duct, and related test connections and manual valves. The isolation dampers are classified as SC1 and Seismic Category B. If radiation is detected, the SCR normal supply and exhaust isolation dampers will shut, the SCR pressurization fans will energize supplying the SCR with filtered outside air.

Tornado dampers close automatically and mitigate the effect of external hazards, as required by REGDOC-2.5.2, Section 7.15.1 (Reference 6.4-1).

## **Radiation Protection**

SCR instrumentation for monitoring of radioactivity is provided in Chapter 7, Section 7.6, and Chapter 12, Subsection 12.3.14.

## **Shielding Design**

Shielding design bases is provided in Chapter 12, Subsection 12.4. Descriptions of the design bases source terms and shielding parameters are presented in Chapter 12, Section 12.4 and Chapter 15, 15.7, respectively. The SCR designated radiation zones are shown in Chapter 12, Figure 12.6-5.

## **Fire Protection**

Fire and smoke detectors, and smoke removal descriptions are provided in Chapter 9A, Section 9A.6.

### **6.4.4 Materials**

Refer to Chapter 9A, Section 9A.5.1 for information pertaining to HVS materials.

### **6.4.5 Interfaces with Other Equipment or Systems**

Interfaces are addressed in Chapter 9A Table 9A.5.1-1, *Reactor Building System Interfaces* and Table 9A.5.2-1, *Control Building Heating, Ventilation and Air-Conditioning System Interfaces*.

### **6.4.6 System and Equipment Operation**

SCR and MCR emergency habitability control room HVS (Chapter 9, Section 9A.5.2) do not operate during normal conditions. The control room HVS maintains the air temperature of the control room habitability envelope within a predetermined temperature range using fan cooling units and air handling units that are supplied with chilled water during normal operations and shutdown. This maintains the SCR emergency habitability function passive heat sink at or below a predetermined temperature.

#### **6.4.6.1 Main Control Room**

Figures 6.4-1 and 6.4-2 show the MCR habitability envelope and location of AHUs, toxic gas monitors, TGFUs, and CRE EFUs located in the CB.

## **High Radiation**

CRE EFUs operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE goes into isolation mode when normal CRE supply and return air dampers close.

## **Toxic Gas**

The MCR remains habitable for the duration of a toxic gas release event.

### **Loss of All Alternating Current**

During a loss of all AC, the MCR remains the control location until habitability temperatures are exceeded or controls and indications are lost due to high temperature effects from I&C and electrical components. The control room operators will relocate to the SCR in the RB if the MCR becomes uninhabitable.

#### **6.4.6.2 Secondary Control Room**

Operation of the emergency habitability portion of the SCR HVAC is automatically initiated on any of the following conditions:

- High radioactivity in the operating RB normal supply AHU outside air intake
- Toxic gas detection in supply duct to the RB
- Loss of all AC power

Operation is also initiated manually (Chapter 9A, Figure 9A.5.1-1 for RB HVAC process flow diagram).

### **High Radiation**

Upon receipt of a high radiation in the operating RB normal supply AHU outside air intake, the normal RB supply and exhaust are isolated from the control room habitability envelope by automatically closing the isolation dampers in the system ductwork. Simultaneously, one of two emergency pressurization fan EFUs automatically starts and begins to deliver filtered air from one of the two unique safety category outside air intake locations.

### **Toxic Gas**

The SCR is not inhabited during a toxic gas release event.

### **Loss of all Alternating Current**

The SCR remains habitable for 72 hours during a loss of all AC. During loss of all AC power, AC power is provided from SC1 batteries.

#### **6.4.7 Instrumentation and Control**

Instrumentation required for SCR habitability design features are provided in Chapter 7, Section 7.3. Details of the radiation monitors used to provide indication of actuation of an isolation and EFU initiation are given in Chapter 12, Subsection 12.3.5.

#### **6.4.8 Monitoring, Inspection, Testing and Maintenance**

Monitoring, inspection, testing, and maintenance are planned and performed to meet system reliability and availability targets. Chapter 9A, Section 9A.5.1 describes HVS inspection and testing.

Control room habitability leak testing is included in the POSAR.

#### **6.4.9 Radiological Aspects**

The radiological dose consequences from DBAs are analyzed and evaluated, and the accompanying results are provided in Chapter 15, Sections 15.5 and results are in Section 15.7.

#### **6.4.10 Performance and Safety Evaluation**

Radiation doses to SCR and MCR personnel are calculated for the accident scenarios where the pressurization system provides filtered air pressurizing the CRH envelope. Radiological dose consequences from normal coolant pipe break outside containment are evaluated in Chapter 15, Section 15.5. There are no DBAs or AOOs that lead to fission product releases.

#### **6.4.11 References**

- 6.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 6.4-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 6.4-3 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 6.4-4 CSA N293-12, "Fire Protection for Nuclear Power Plants," CSA Group.
- 6.4-5 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

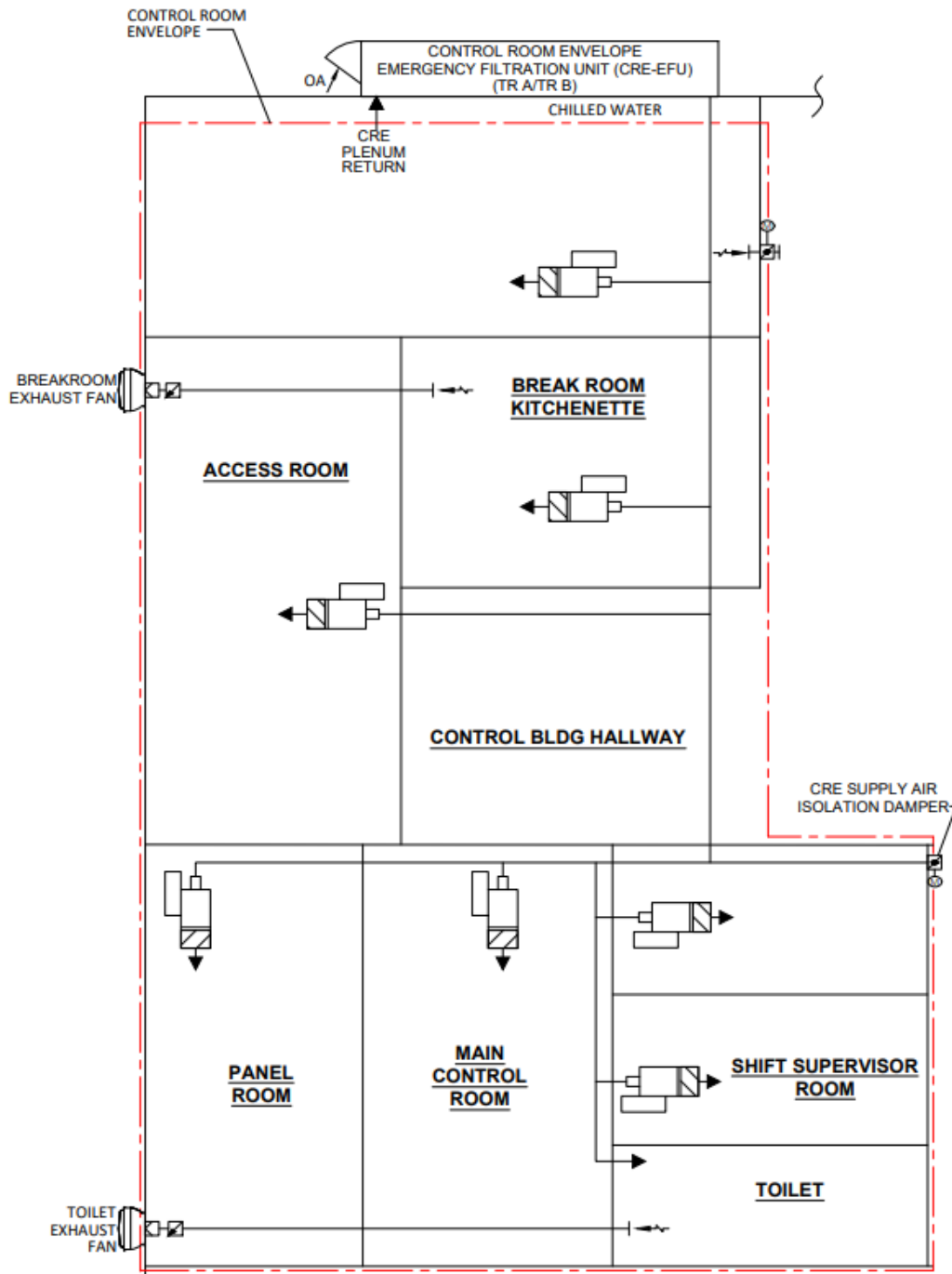


Figure 6.4-1: Main Control Room Habitability Envelope

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

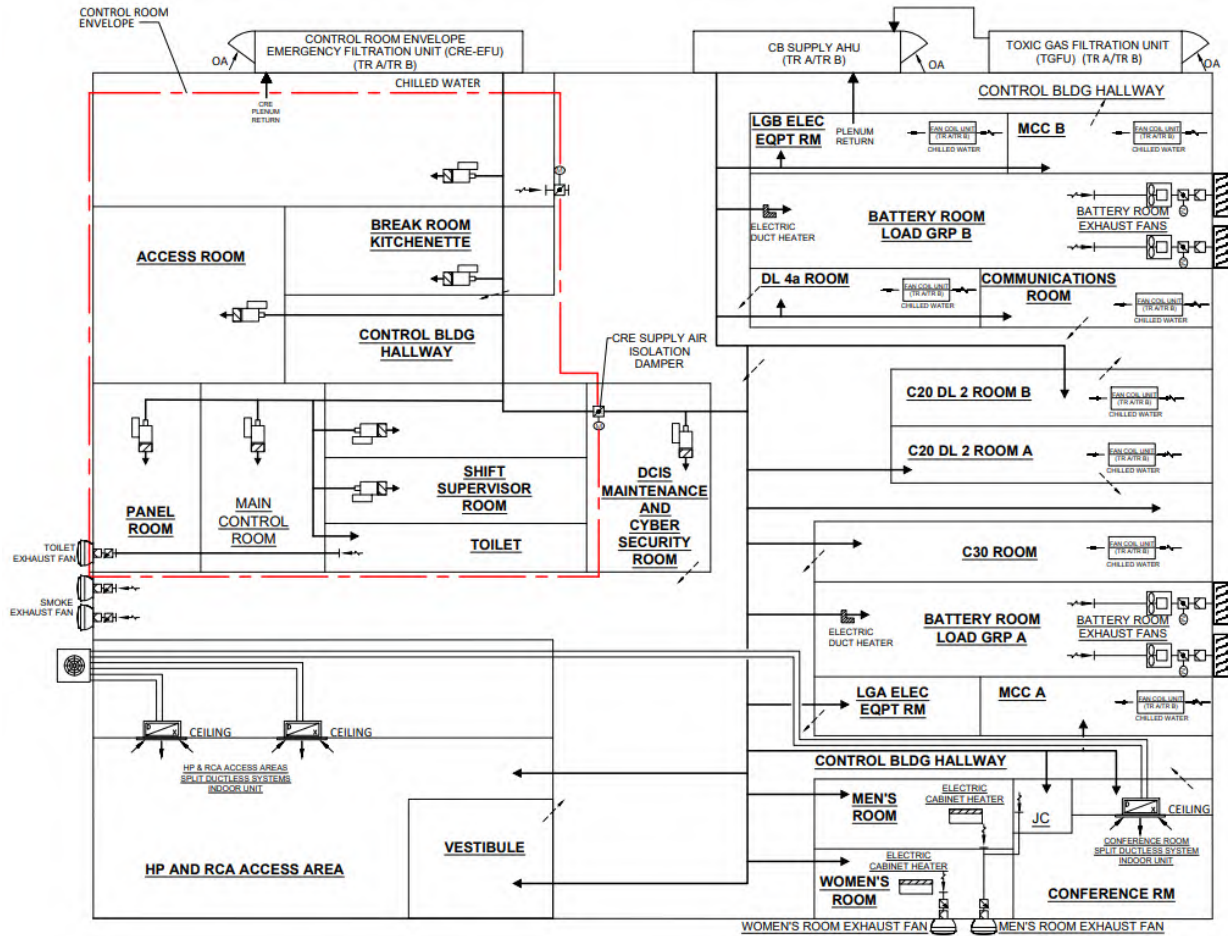


Figure 6.4-2: Control Building HVAC Simplified Diagram

NEDO-33955 REVISION 0  
NON-PROPRIETARY INFORMATION

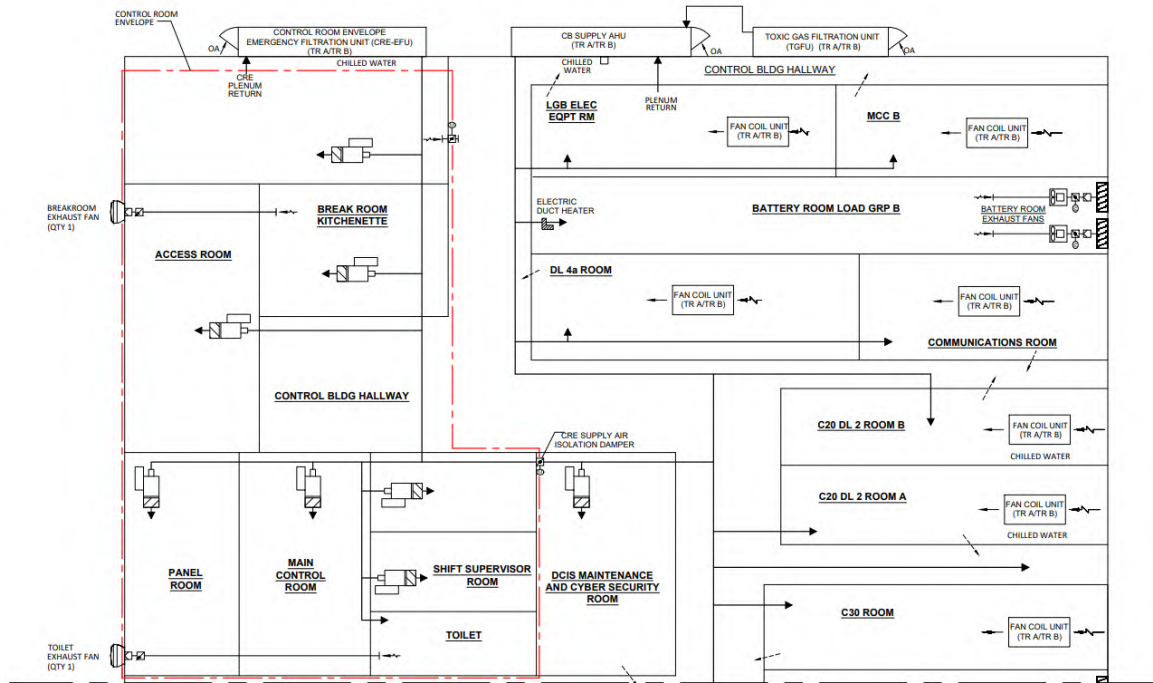


Figure 6.4-3: Control Building HVAC Simplified Diagram

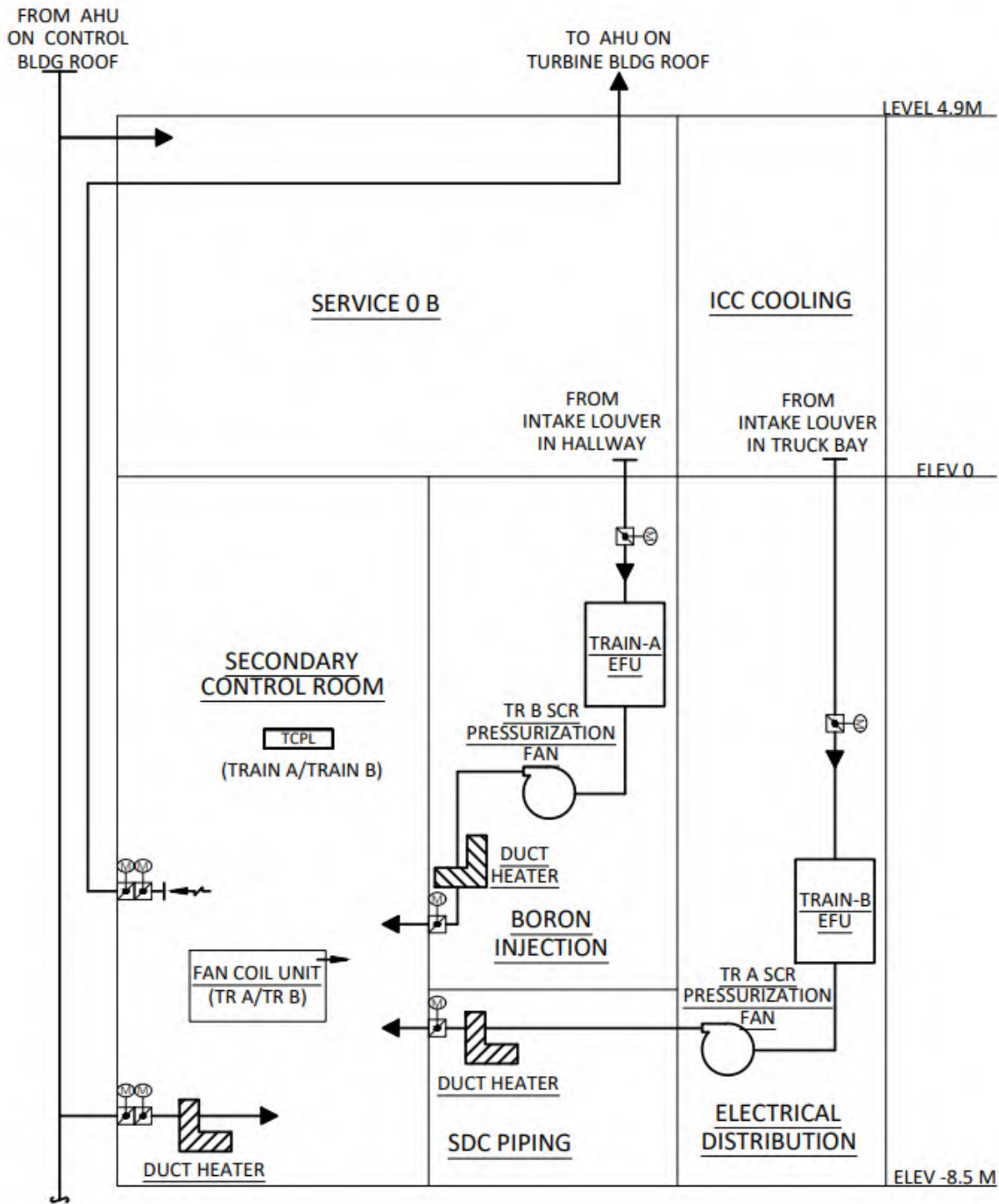


Figure 6.4-4: Reactor Building Secondary Control Room





**HITACHI**

GE Hitachi Nuclear Energy

NEDO-33956

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 7  
Instrumentation and Control**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
3D	Three-Dimensional
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Automatic Rod Insertion
ARMS	Area Radiation Monitoring Subsystem
ATLM	Automatic Thermal Limit Monitor
ATS	Anticipatory Trip System
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DL	Defense Line
DEC	Design Extension Condition
DPS	Diverse Protection System
EOC	Emergency Operations Centre
EOP	Emergency Operating Procedure
ERF	Emergency Response Facility
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
FMCRD	Fine Motion Control Rod Drive
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWRIV	Feedwater Reactor Isolation Valve
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HFE	Human Factors Engineering
HSI	Human-System Interface

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
LAN	Local Area Network
LPRM	Lower Power Range Monitor
MCR	Main Control Room
MRBM	Multi-Channel Rod Block Monitor
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MSRIV	Main Steam Reactor Isolation Valve
OLC	Operational Limits and Conditions
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring System
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
SAMG	Severe Accident Management Guideline
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCR	Secondary Control Room
SCRR	Selected Control Rod Run-In
SDC	Shutdown Cooling System
SPDS	Safety Parameter Display System
SSC	Structures, Systems, and Components
STP	Simulated Thermal Power

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
TMR	Triple Modular Redundant
UPS	Uninterruptible Power Supply
VDU	Visual Display Unit
WRNM	Wide-Range Neutron Monitor

## TABLE OF CONTENTS

<b>7.0</b>	<b>INSTRUMENTATION AND CONTROL .....</b>	<b>7-1</b>
7.1	Instrumentation and Control Introduction and Overview.....	7-1
7.1.1	Relationship Between Instrumentation and Control Functions and Plant-Level Defense Lines .....	7-1
7.1.2	Instrumentation and Control System Classification .....	7-2
7.1.3	Regulatory Requirements and Industry Standards.....	7-3
7.1.4	References .....	7-4
7.2	Instrumentation and Control System of Systems .....	7-7
7.2.1	Distributed Control and Information System.....	7-7
7.2.2	Unit Data Highway Network .....	7-8
7.2.3	Network Managed Switches .....	7-8
7.2.4	Plant Data Highway Network .....	7-9
7.2.5	References .....	7-10
7.3	Distributed Control and Information System Functions .....	7-14
7.3.1	System C10 – Defense Line 3 / Safety Class 1.....	7-14
7.3.2	System C20 – Defense Line 4a / Safety Class 2.....	7-26
7.3.3	System C20 – Defense Line 2 / Safety Class 3 / Nuclear Segment .....	7-33
7.3.4	System C30 – Non-Safety Class / Balance of Plant Segment.....	7-47
7.3.5	References .....	7-53
7.4	Digital Instrumentation and Control System Development Processes.....	7-85
7.4.1	Design Control.....	7-85
7.4.2	Defence-in-Depth and Architecture Design .....	7-85
7.4.3	Instrumentation and Control System Life Cycle .....	7-87
7.4.4	Cyber Security Life Cycle.....	7-87
7.4.5	Compliance Alignment.....	7-88
7.4.6	References .....	7-88
7.5	Instrumentation and Control in the Main Control Room.....	7-95
7.5.1	Main Control Room Use.....	7-95
7.5.2	Main Control Room Layout .....	7-96
7.5.3	References .....	7-97
7.6	Instrumentation and Control in Secondary Control Room .....	7-99
7.6.1	Control Transfer Function .....	7-99
7.6.2	Secondary Control Room Use .....	7-99

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

7.6.3	Secondary Control Room Layout.....	7-99
7.6.4	References .....	7-100
7.7	Instrumentation and Control in the Emergency Response Facilities .....	7-101
7.7.1	References .....	7-101
7.8	Hazard Analysis for Instrumentation and Control Systems .....	7-102
7.8.1	References .....	7-102



## LIST OF TABLES

Table 7.1-1: I&C System and Equipment Standards .....	7-5
Table 7.3-1: C10 SC1 I&C Functions and Initiation Signals .....	7-56
Table 7.3-2: BWRX-300 Candidate Accident Monitoring Variables .....	7-58
Table 7.3-3: DL4a Diverse Protection System Functions and Initiating Signals .....	7-64
Table 7.3-4: DL2 Anticipatory Trip and Block Functions and Initiating Signals .....	7-66
Table 7.3-5: C10 SC1 I&C Compliance Alignment .....	7-67
Table 7.3-6: C20 Defense Line 4a Compliance Alignment .....	7-69
Table 7.3-7: C20 Defense Line 2 Nuclear Segment Compliance Alignment .....	7-70
Table 7.3-8: C30 BOP Segment Compliance Alignment .....	7-71
Table 7.4-1: Digital I&C System Development Process Compliance Alignment.....	7-89

## LIST OF FIGURES

Figure 7.1-1: DCIS Defense Lines and Classifications .....	7-6
Figure 7.2-1: BWRX-300 DCIS Network Architecture.....	7-11
Figure 7.2-2: Typical I&C Controller and Component Network Connections.....	7-12
Figure 7.2-3: Rapid Spanning Tree Network Managed Switches .....	7-13
Figure 7.3-1: Reactor Mode Switch .....	7-72
Figure 7.3-2: DL3 Safety Class 1 Functional Architecture .....	7-73
Figure 7.3-3: BWRX-300 DL3 Actuation Logic .....	7-74
Figure 7.3-4: BWRX-300 Hydraulic Scram Solenoid Control and Power .....	7-75
Figure 7.3-5: BWRX-300 Isolation Valve Solenoid Control and Power .....	7-76
Figure 7.3-6: BWRX-300 Core Map with Monitoring Locations and FMCRD Power Assignments .....	7-77
Figure 7.3-7: DL3 Power Range Neutron Monitoring Block Diagram.....	7-78
Figure 7.3-8: DL4a Safety Class 2 Functional Architecture .....	7-79
Figure 7.3-9: DL4a Analog Signal Splitters.....	7-80
Figure 7.3-10: Representative Diverse Protection System Scram Block Diagram .....	7-81
Figure 7.3-11: DL2 Safety Class 3 Nuclear Segment Functional Architecture .....	7-82
Figure 7.3-12: Example Anticipatory Trip System Scram Block Diagram.....	7-83
Figure 7.3-13: C30 Non-Safety Class BOP Functional Architecture .....	7-84
Figure 7.4-1: BWRX-300 I&C Architecture Design Process .....	7-90
Figure 7.4-2: Overall BWRX-300 I&C System Life Cycle.....	7-91
Figure 7.4-3: Software Related Activities in I&C System Life Cycle.....	7-92
Figure 7.4-4: Cyber Security Life Cycle .....	7-93
Figure 7.4-5: Cyber Security Architecture Security Boundaries .....	7-94
Figure 7.5-1: MCR and Surrounding Areas (Top View) .....	7-98

## **7.0 INSTRUMENTATION AND CONTROL**

The Instrumentation and Control (I&C) systems for the BWRX-300 are in fact a system of systems. The information in Chapter 7 is organized to systematically present the I&C design bases in the necessary context to support an understanding of the individual I&C system designs and safety features. Information on the relationship of the I&C systems to plant-level requirements is presented in Section 7.1. Next, information on the architecture of the I&C system of systems is presented in Section 7.2. Then information on the individual I&C systems, including accident monitoring instrumentation, is presented in Section 7.3. The development processes for the digital I&C systems are presented in Section 7.4. The I&C systems in the Main Control Room (MCR) are described in Section 7.5. The I&C systems in the Secondary Control Room (SCR) are described in Section 7.6. The I&C systems in the Emergency Response Facilities are described in Section 7.7. Hazards analyses for the BWRX-300 I&C systems are described in Section 7.8.

### **7.1 Instrumentation and Control Introduction and Overview**

The BWRX-300 Instrumentation and Control (I&C) systems are designed to support the plant safety strategy described in Chapter 3, Section 3.1.

#### **7.1.1 Relationship Between Instrumentation and Control Functions and Plant-Level Defense Lines**

The I&C functions are allocated to the various defense lines.

Defense Line 1 (DL1) includes the quality measures taken to minimize potential for failures and initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defence. These quality measures cover the design, construction, operation, and maintenance of the plant. DL1 also includes the use of conservatism in design and analyses. Accident monitoring instrumentation supports functions in more than one DL and receives DL1 treatment by application of applicable industry standards for accident monitoring instrumentation, as shown in Table 7.1-1.

Defense Line 2 (DL2) contains plant functions (e.g., anticipatory reactor trips) designed to control or initiate responses to Postulated Initiating Events (PIE), especially Anticipated Operational Occurrences (AOO), before any parameters reach a Defense Line 3 (DL3) actuation setpoint. Functions that normally operate to actively control reactor parameters are also part of DL2 (e.g., blocking control rod withdrawal).

DL3 contains plant functions that act to mitigate a PIE (i.e., AOO or Design Basis Accidents (DBA)) by preventing fuel damage when possible, assuring the integrity of the barriers to release, and placing the plant in a safe state. DL3 also includes functions credited to maintain the plant in a safe condition following mitigation of PIE, until normal operations are resumed. Primary DL3 functions include reactor scram, Reactor Pressure Vessel (RPV) and containment isolations, and Isolation Condenser System (ICS) to mitigate the consequences of AOOs or DBAs. DL3 functions are needed when DL2 is not effective at intercepting a PIE or when a PIE is simply beyond the capabilities of the DL2 functions.

DL3 is credited to mitigate PIEs independent of DL2 and DL4a and is therefore required to be independent from DL2 and DL4a to the extent practicable. GEH defines practicable in this context as possible. Exceptions related to the I&C systems are cases where probabilistic safety analyses indicate safety is improved by the sharing of components (e.g., including hydraulic scram in DL2, DL3, and DL4a and using the DL4a Fine Motion Control Rod Drives (FMCRD) Fast Motor Run-in function to back up the hydraulic scram).

DL4 comprises two subsets of functions, designated as Defense Line 4a (DL4a) and Defense Line 4b (DL4b) functions. DL4a functions place and maintain the plant in a safe state in case of PIE with failure of the DL3 functions (i.e., Design Extension Conditions (DEC) without core damage). The need for DL4a functions generally arises when specific, postulated Common Cause Failures (CCF) occurring in DL3. The DL4a functions will actuate for any condition that satisfies the actuation criteria logic.

DL4a is used to mitigate PIEs that are not mitigated by DL2 (e.g., because the baseline plant response would be the actuation of DL3 functions – typically DBAs) and provides a second credited defense line for such PIEs (after DL3). DL4a is designed to work in tandem with DL2 to ensure all AOOs and DBAs resulting from a single failure are mitigated by two defense lines among DL2, DL3, and DL4a. Accordingly, DL4a is not required to be independent and diverse from DL2. DL4a can be used, along with unaffected DL2 functions, to mitigate a PIE as part of the same event sequence (i.e., to act as a single defense line and not as two independent defense lines in an DEC analysis). All AOOs and DBAs resulting from a single failure are required to be mitigated by DL3 and separately by DL2, DL4a, or a combination of DL2 and DL4a. DL2 and DL4a are not credited to mitigate the same PIEs independently of each other and are therefore not required to be independent from each other.

DL4b provides protection for extreme events, multiple (i.e., combined) events, or multiple failures that defeat DL2, DL3, and DL4a. DL4b functions prevent or mitigate a severe accident while keeping radioactive releases to acceptable levels. DL4b is required to be independent of DL2, DL3, and DL4a to the extent practicable, with exceptions documented and justified.

The DL independence requirements are consistent with the defence-in-depth strategy (i.e., the crediting of defense lines in the Fault Evaluation and Deterministic Safety Analyses).

CNSC REGDOC-3.6, "Glossary of CNSC Terminology," (Reference 7.1-1) defines practicable as "Technically feasible and justifiable while taking cost-benefit considerations into account." Although not specifically required by the BWRX-300 Safety Strategy, there is independence between DL2 and DL4a to a level that does meet this definition of practicable; it is not justifiable or cost-effective to require full independence between DL2 and DL4a (i.e., the type of independence required between DL3 and DL2/DL4a). The only aspects of independence required between equipment performing DL2 and DL4a functions are electrical isolation and physical separation. These interface requirements stem from the different safety classifications of the equipment.

Defense Line 5 (DL5) includes emergency preparedness measures to cope with potential unacceptable releases in case the first four DLs are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation. DL5 is supported by the accident monitoring instrumentation.

### **7.1.2 Instrumentation and Control System Classification**

The BWRX-300 Distributed Control and Information System (DCIS) is an integrated control and monitoring system for the power plant. The DCIS is arranged in three Safety Classified DCIS segments and a Non-Safety Class segment with appropriate levels of hardware and software quality corresponding to the system functions they control and their DL location, as described in Chapter 3, Section 3.2. The DCIS provides control, monitoring, alarming and recording functions. The various bus segments of the integrated DCIS are designed to operate autonomously.

The DL3 functions are assigned to Safety Category 1 and performed by Safety Class 1 (SC1) equipment, except with the following exceptions:

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

1. Structures, Systems, and Components (SSC) are only needed after the first 24 hours of the event are classified as Safety Class 2 (SC2), and
2. SSC are only needed after the first 7 days of the event are classified as Safety Class 3 (SC3).

The DL4a functions are assigned to Safety Category 2 and performed by at least SC2 equipment, except for SSC that are only needed after the first 7 days of the event, which are classified as SC3. DL4a functions complete the safety objective via an independent and diverse means of logic and actuation if the DL3 function is not completed due to a C10 failure.

The DL2 functions are assigned to Safety Category 3 and performed by at least SC3 equipment. The safety functions in DL2 must be performed independently from diverse DL3 functions providing protection for the same event.

The DL4b functions are categorized as Safety Category 3 and generally performed by at least SC3 equipment unless other requirements are justified.

The Safety Category 1 DCIS functions are allocated to the C10 system that requires minimal support equipment (e.g., no active systems such as Heating, Ventilation and Air Conditioning (HVAC) or cooling water for at least 72 hours). Safety Category 1 DCIS functions are implemented on C10 SC1 I&C system hardware and software platforms located in three separate divisional fire barrier rooms in the Reactor Building. The DL4a (i.e., Safety Category 2) and DL2 (i.e., Safety Category 3) DCIS functions are implemented in different subsystems of C20. The DL4a DCIS functions are implemented with SC2 equipment. The majority of the C20 SC2 equipment is located in its own fire barrier room in the Control Building with the remainder located in compartmentalized fire barrier rooms in the Reactor Building. The DL2 functions are implemented with SC3 equipment. The C20 SC3 equipment is located in two separate fire barrier rooms in the Control Building.

The C30 controllers for the Non-Safety Class equipment are designed to prevent random I&C component failures from initiating plant transients. Vendor supplied Non-Safety Class functions are input through a C30 gateway. The C30 DCIS functions can be implemented on vendor packages (or skids) located in a separate fire barrier room in the Control Building.

The system designations of C10, C20, and C30 represent an initial decomposition of the overall I&C systems based on safety classification. Further refinement of the system decomposition, based on functional grouping within a safety class and equipment selection, will be performed as the I&C Architecture design process progresses. This process is described in Subsection 7.4.2.2.

The BWRX-300 I&C DLs and classifications are shown in Figure 7.1-1.

### **7.1.3 Regulatory Requirements and Industry Standards**

The BWRX-300 I&C systems also comply with CSA N290.14:15 (Reference 7.1-2) using the International Electrotechnical Commission (IEC) 61513 (Reference 7.1-3) option in Table 1. The BWRX-300 I&C does not rely on IEC 61226:2009 (Reference 7.1-4) for compliance with CSA N290.14:15. Instead, the BWRX-300 uses Section 7 of IEC 61226:2020 (Reference 7.1-5) to identify applicable I&C equipment standards. A summary of the key system and equipment standards applied to the I&C systems based on IEC 61226:2020 is shown in Table 7.1-1. The overall I&C regulatory conformance plan is documented in a BWRX-300 Plant I&C Systems Nuclear Regulations and Standards Conformance Plan. The BWRX-300 I&C systems are based on the requirements of Canadian Nuclear Safety Commission (CNSC) REGDOC-2.5.2 (Reference 7.1-6), and CNSC REGDOC-1.1.2 (Reference 7.1-7).

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Section 7.3 provides a summary of the alignment for each I&C system to the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513, Institute of Electrical and Electronics Engineers (IEEE) Std 603 (Reference 7.1-8), and IEC 63147 (Reference 7.1-9). Section 7.4 provides a summary of the alignment for the I&C system development process to the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603.

**7.1.4 References**

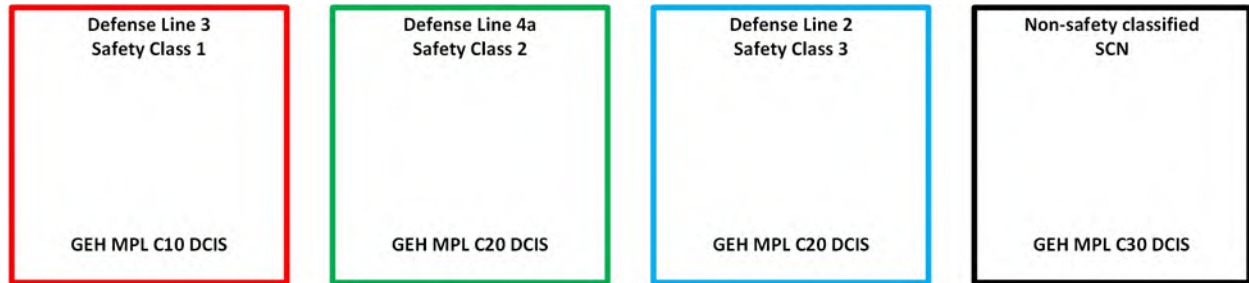
- 7.1-1 CNSC Regulatory Document REGDOC-3.6, "Glossary of CNSC Terminology."
- 7.1-2 CSA N290.14:15, "Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants", CSA Group.
- 7.1-3 IEC 61513, "Nuclear power plants – Instrumentation and control important to safety – General requirements for systems," International Electrotechnical Commission.
- 7.1-4 IEC 61226, "Nuclear power plants – Instrumentation, control and electrical power systems important to safety – Categorization of functions and classification of systems," International Electrotechnical Commission, July 2009.
- 7.1-5 IEC 61226, "Nuclear power plants – Instrumentation, control and electrical power systems important to safety – Categorization of functions and classification of systems," International Electrotechnical Commission, April 2020.
- 7.1-6 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 7.1-7 CNSC Regulatory Document REGDOC-1.1.2, "License Application Guide: License to Construct a Reactor Facility."
- 7.1-8 IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 7.1-9 IEC 63147, "Criteria for accident monitoring instrumentation for nuclear power generating stations," International Electrotechnical Commission.
- 7.1-10 IEC 61508, "Functional safety of electrical/electronic/programmable electronic safety-related systems – Parts 1 to 7," Commented version, International Electrotechnical Commission.

**Table 7.1-1: I&C System and Equipment Standards**

<b>Safety Class</b>	<b>Description</b>	<b>Systems</b>	<b>Equipment</b>
SC1, SC2, and SC3	I&C Equipment supporting Safety Category 1, 2 and 3 functions (systems C10 and C20)	IEC 61513 (I&C architecture and general system requirements) IEC 60709 (separation) IEC 63147 (accident monitoring) IEC 60812 (FMEA)	IEC 61000-4, IEC 61000-6-2 (electromagnetic compatibility)
SC1	I&C Equipment supporting Safety Category 1 functions (system C10)	IEC 60880 (Category A software) IEC 60987 (hardware)	IEC 60780 (environmental) IEC 60980 (seismic) IEC 61500 (network communication)
SC2	I&C Equipment supporting Safety Category 2 functions (system C20)	IEC 60987 (hardware) IEC 62138 (Category B software)	IEC 60780 (environmental) IEC-61508 (Reference 7.1-10)
SC3	I&C Equipment supporting Safety Category 3 functions (system C20)	IEC 62138 (Category C software)	IEC 60980 (seismic) IEC 61508

(1) The identified software standards satisfy CSA N290.14:15

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



*NOTE: DL4b is not part of the DCIS.*

**Figure 7.1-1: DCIS Defense Lines and Classifications**



## **7.2 Instrumentation and Control System of Systems**

The three DCIS systems (i.e., C10, C20, and C30) are developed and integrated into the Defence-in-Depth (D-in-D) architecture, as described in Subsection 7.4.2. The DCIS systems are networked using a managed network switch scheme that allows the parts to operate independently and prevent faults from adversely affecting each other while appearing seamless to the plant operator. The BWRX-300 I&C segments are interconnected (networked) to support common services like alarming, Visual Display Units (VDU), recording and sending data to the Emergency Response Facilities while maintaining appropriate separation between the segments and providing for the necessary cyber security.

The BWRX-300 I&C systems architecture incorporates defensive cyber security features designed to meet regulatory requirements, as described in Subsection 7.4.4. Managed network switches are standard industrial components with a proven record for reliable and safe operation in similar network applications. The BWRX-300 cyber security architecture provides cyber security defensive levels. Data flow and communications between defence levels are managed by boundary devices that implement unidirectional flow where required to prevent a lower level from compromising a higher level. Isolation features that are provided at interfaces between systems with different classification levels are classified to the higher classification. Signals used from lower Safety Class equipment are not used to perform higher Safety Category functions in the higher Safety Class equipment.

### **7.2.1 Distributed Control and Information System**

The various DCIS systems are implemented on different hardware and software platforms appropriate to the safety classification of the functions they are performing, as indicated by the following colors in Figure 7.2-1:

- Red – C10, DL3 and SC1
- Green – C20, DL4a and SC2
- Blue – C20, DL2 and SC3
- Black – C30, No DL and Non-Safety Class
- Yellow – Enterprise network

The SC1 DCIS (C10) is completely isolated from the rest of the BWRX-300 DCIS except for the optically isolated data links provided through unidirectional boundary devices to SC3 (C20). The VDUs dedicated to C10 are the only mechanism by which the SC1 systems can be controlled. They can also be monitored by the C10 VDUs and be monitored, alarmed, and recorded in C20. The SC1 DCIS is divided into three divisions and those divisions are separately powered by the three divisions of the R10 electrical system. The three SC1 DCIS divisions operate completely independently from each other (except for voting) and from the lower safety class DCIS. Isolated information only flows in one direction through boundary devices to the SC3 networks so that SC1 system information may be alarmed, monitored, and recorded (but not controlled) with SC2 and SC3 equipment (e.g., VDUs, printers, alarm system, and historians).

C20 is divided into two DLs. One is DL4a and performs Safety Category 2 functions using SC2 equipment. The other is DL2 and performs Safety Category 3 functions using SC3 equipment. DL4a actuation logic operates on an analog-based I&C platform. It communicates status and diagnostic information to the DL2 portion of C20, which is based on digital I&C technology. The C20 DCIS is redundant and segmented into two parts. The two network segments are the Nuclear Segment and the Balance of Plant (BOP) Segment. The two bus segments are classified as SC3. The Nuclear Segment uses controllers classified as SC3. The BOP Segment interfaces with C30

and Non-Safety Class controllers to acquire information to support human-system interfaces (HSI) integration in C20. Both segments have their own redundant Unit Data Highway, and each independently provides the control and monitoring capability of nuclear and BOP I&C for their segment. Each segment has their own redundant network managed switches classified as SC3. Each segment can operate independently of the other (although it appears seamless to the operator). Each segment has associated VDUs, historians, and a common alarm system. The C20 segments use Triple Modular Redundant (TMR) and redundant controller architectures for various functions to support reliability goals or prevent adverse actuations for anticipated hardware component failures.

The Nuclear Segment supports the SC3 controllers (which include the major reactor control systems) by acquiring DL3 data from the C10, SC1 controllers via unidirectional boundary devices. It also acquires DL4a diagnostics.

The BOP Segment provides for control of the Non-Safety Class plant functions and controllers used mainly for power generation. It also provides gateways to interface non-native controllers to the BOP network segment. Although lower classified equipment, these functions use TMR controller architectures (referred to hereafter as TMR controllers) to prevent random I&C component failures from causing plant transients.

The I&C for DL4b is not implemented through the BWRX-300 DCIS.

### **7.2.2 Unit Data Highway Network**

The C20 and C30 DCIS components are connected to the two Unit Data Highway networks shown in Figure 7.2-1 using the general switch arrangement shown in Figure 7.2-2. The components are “dual ported,” specifically they are each connected to separate but redundant network switches. The switches are discussed in the next section, but the important reliability point is that either connection to the network for the component and either network switch provides full functionality; failovers are seamless. The controllers are either dual redundant or TMR and the connections and switches are dual redundant. The design is single-failure proof and components, and connections are monitored and alarmed. The various components can be replaced online without affecting either safety or power generation.

Because of the network managed switches, the network segmentation appears transparent to the operator in normal operation (i.e., any C20 VDU can monitor or control any C20 or C30 function). However, in the unlikely event either of the individual redundant network segments becomes unavailable, the remaining segment remains operational.

### **7.2.3 Network Managed Switches**

The blue lines in Figure 7.2-1 indicate that the redundant C20 networks appear as busses; in fact, they are a rapid spanning tree network of managed Ethernet switches as shown on Figure 7.2-3.

Each managed network switch provides standard Ethernet switch capability but also provides additional cyber security advantages because they have security features. These include identification of authorized equipment addresses (locking out unauthorized equipment even if it is connected to the network), the capability to ignore or not uplink (to other segments), the capability to control which nodes are allowed to communicate and the capability to alarm abnormal network traffic. Each switch is programmed to allow intercommunication only between defined nodes and to refuse communication to or from a node whose identification has not been previously defined (i.e., a “stranger” plugging into the network). Only when a switch determines that an information data packet is destined for a node on another switch (and that communication is allowed) is the information put on an uplink to another switch. The network switches learn and maintain their own forwarding tables containing a list of the nodes and hosts on their respective network

segment. When a network switch receives a data communication packet, it forwards only that data packet to the segment to which the receiving host is connected. These protocols prevent data traffic between devices on the network from affecting devices on other segments of the network.

The uplink ports on the switches are connected radially and in a data communication ring because multiple interconnections increase reliability. Specifically, the switches use a “rapid spanning tree protocol” to automatically enable and disable ports so there is only one path from the nodes of one switch to another. Should a path become disabled, the switches automatically reconfigure to establish another path through the remaining switches and fiber optic cable paths. Reconfiguration requires no operator input and is usually accomplished in much less than a second.

The loss of one switch per segment has no effect on plant operation or data. The failure is indicated and is repairable with the system online. In the highly unlikely event of both switches of a segment simultaneously failing, that segment is lost. However, the remaining segments are unaffected and individual nodes connected to the failed switches can continue to function. The remaining switches then automatically reconfigure their uplink ports such that the remaining segments automatically find available data paths between each other. No switch or network failure can adversely affect DL3 / Safety Category 1 functions because DL3 is isolated from these networks by unidirectional boundary devices.

As described in the previous section, each switch “node” (workstation, display, and controller) is connected to redundant switches of the segment. These connections support normal plant operation. Each switch has redundant power feeds and can work from either power source. The switches and connected controllers support extensive component and data self-diagnostics, and failures are indicated. Finally, the switches are connected to a cyber security monitoring system that provides alarms for intrusion detection (or more likely, intrusion attempts) even if the switches and the network remain unaffected. Additionally, each switch has the capability to monitor, control, reject, and report unexpected and excessive (“data storm”) traffic on its respective network segment.

The BWRX-300 I&C network is not a single network and therefore network failures cannot be the result of a single failure. The networks are redundant and segmented to support the I&C systems with high reliability. The BWRX-300 I&C networks are highly fault tolerant no single network failure can adversely affect plant operation.

#### **7.2.4 Plant Data Highway Network**

Figure 7.2-1 shows the Plant Data Highway. The communications on the nuclear and BOP Unit Data Highways are rigorously controlled and monitored; the managed switches control the information sent to the Plant Data Highway managed switches forming the Plant Data Highway. The Plant Data Highway is used for important but non-essential and non-control services like printers, plotters, long-term data storage (the current requirement is to keep plant data for at least two years). Information sent off-site is through the unidirectional boundary devices, the intent is to send plant data (typically once per second) through the unidirectional boundary device to the server. Various lower cyber security levels of users are connected to the server, including the on-site or off-site Emergency Response Facilities and utility business or engineering networks that may have their own cyber security restrictions. Only the server can respond to off-site requests for information and the only information it has is the data sent to it through the unidirectional boundary devices. No plant DCIS component receives or has to respond to or is able to respond to an off-site data request. No off-site component can “reach” the Plant Data Highway and the network managed switches would not allow any such communication to reach the Plant Data Highway or C20 and C30 Unit Data Highways.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Access to plant cyber essential assets is appropriately protected and any external or business networks in the plant, including the control rooms, is always physically “air gapped” with access control such that they cannot be inadvertently connected to the plant DCIS networks even though the network managed switches would support such a connection.

**7.2.5 References**

None.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

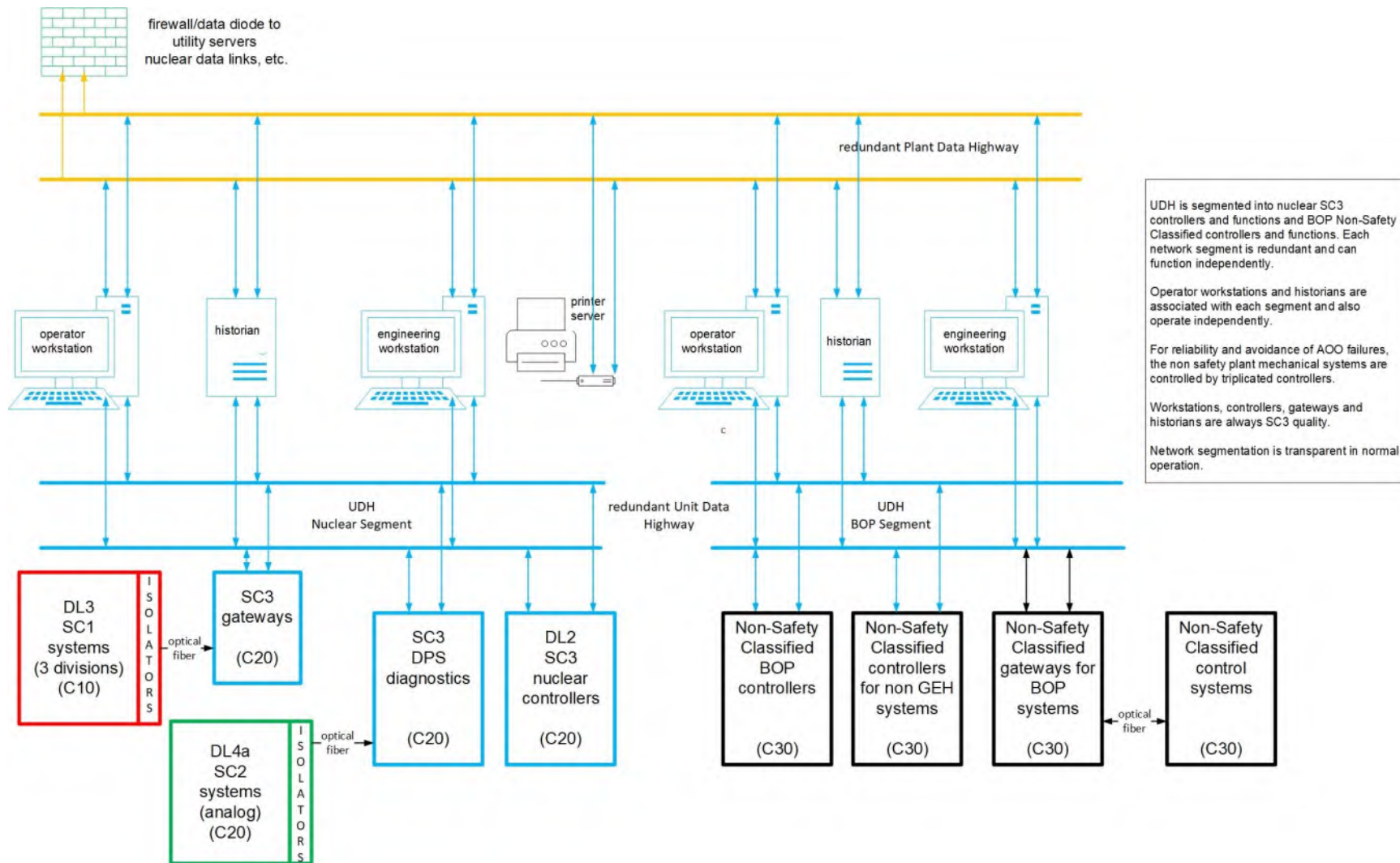
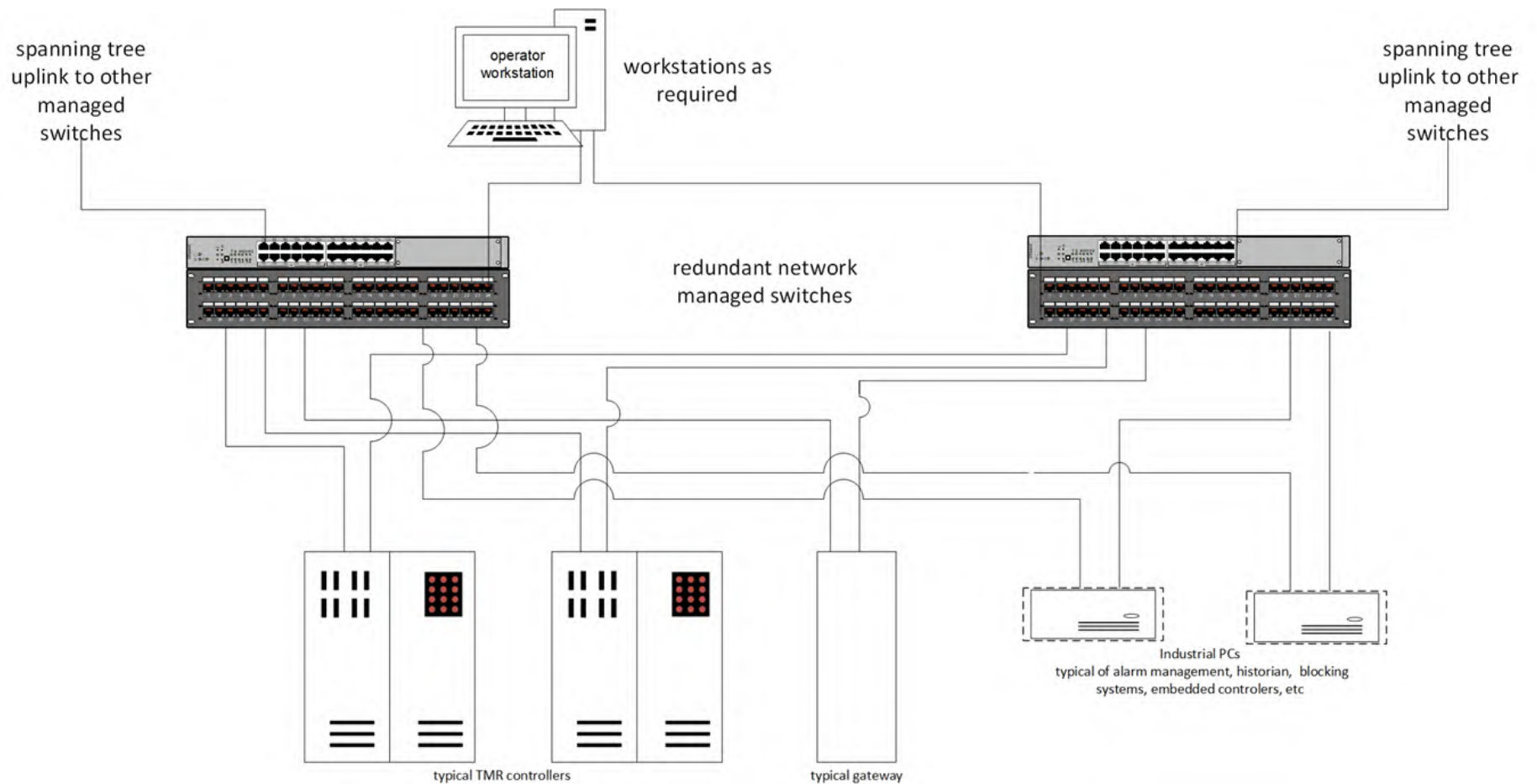


Figure 7.2-1: BWRX-300 DCIS Network Architecture

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.2-2: Typical I&C Controller and Component Network Connections**

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Redundant network UDH nuclear and BOP segments and PDH are actually pairs of network managed switches.

Most equipment and all controllers, operator workstations and historians are dual ported, one port to each switch.

Any redundant switch can be lost without affecting plant operation.

Any segment can be lost without affecting the other segments.

Managed switch uplinks (inter-switch communication) normally allow any segment controller, historian or workstation to send/receive data to any other.

Dark black links indicate normal switch communication, the lighter black links represent possible connections should ports fail or be shut down for self diagnostic monitoring reasons.

Managed switches control which nodes are allowed to communicate and can prevent any unauthorized internal or external communication.

Managed switches have dual power supplies.

Conceptual Cyber Security monitoring connections shown. Other cyber security configurations are possible.

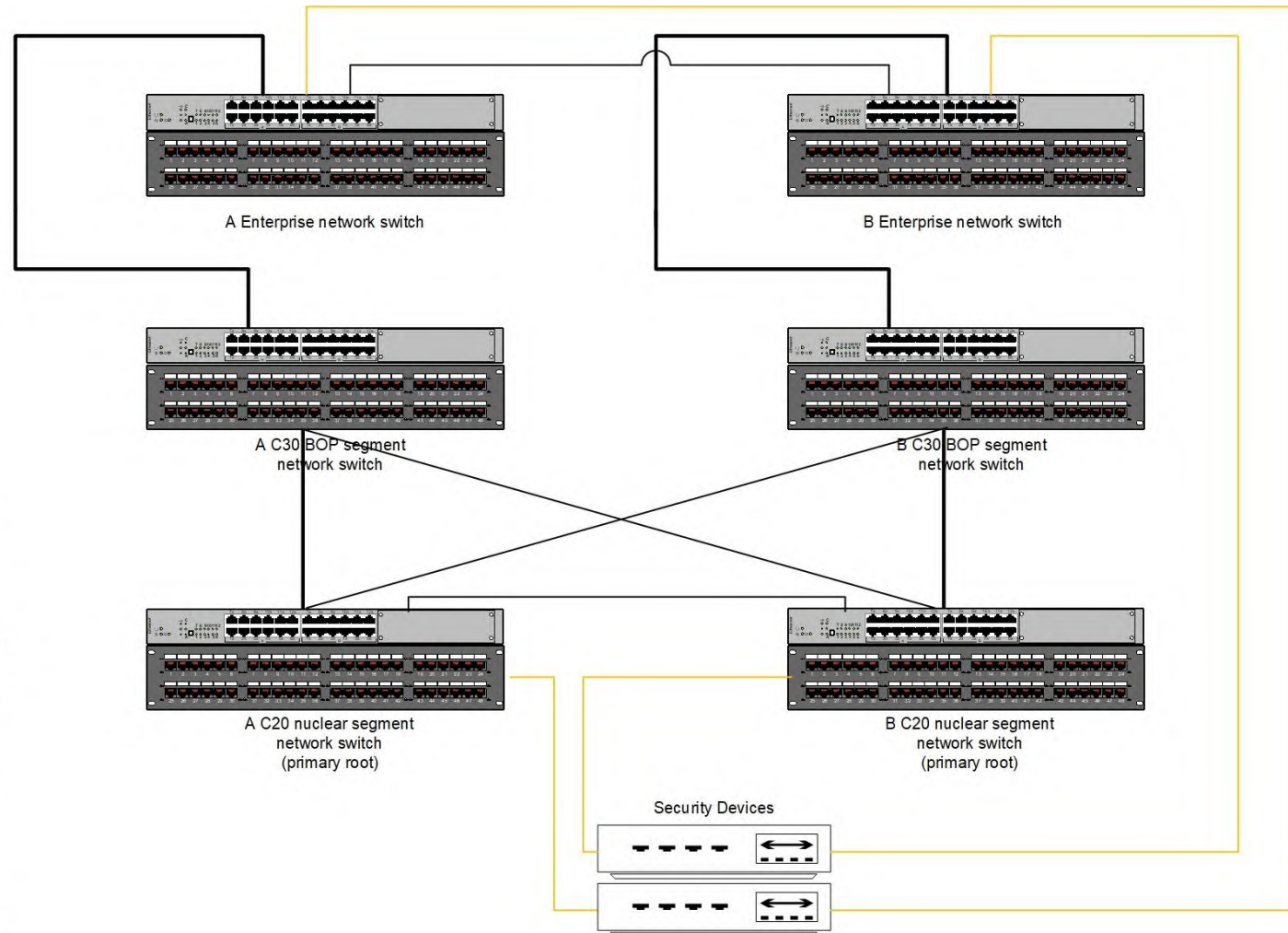


Figure 7.2-3: Rapid Spanning Tree Network Managed Switches



### 7.3 Distributed Control and Information System Functions

The Reactor Mode Switch is an input to the DCIS that inserts or removes operating bypasses for the actuation functions. It is common to the DLs. The Reactor Mode Switch is classified as a C10 SC1 component. The Reactor Mode Switch is a four-position switch with the following positions:

- RUN
- STARTUP
- SHUTDOWN
- REFUEL

The switch is manually positioned by the operator as the reactor is transitioned from cold (refueling outage) to rated power. The switch position determines and provides information to the various controllers needing the Reactor Mode Switch position information to provide the correct interlocks for system operation. As an example, the DL3 logic requires the Reactor Mode Switch position to be in STARTUP to enable a scram at ~15% power and in the RUN position to enable the high power flux scram. DL2 logic requires the Reactor Mode Switch to be in STARTUP to enable the Wide-Range Neutron Monitoring (WRNM) short period anticipatory scram and to disable the short period anticipatory scram in the RUN mode.

A block diagram of the Reactor Mode Switch is shown in Figure 7.3-1. The Reactor Mode Switch is located in the MCR to support normal plant operation. The Reactor Mode Switch is not needed to support any of the functions performed in the SCR.

The switch provides four positions to three divisions of C10 SC1 I&C and isolated signals to an additional four separate systems via separate section of a rotary switch. Each section is isolated from the other sections:

- C10 Three Divisions (DL3 / SC1)
- C20 Diverse Protection System (DPS) (DL4a / SC2)
- C20 Anticipatory Trip System (ATS) DL2 / SC3)
- C20 Automatic Power Regulator DL2 / SC3)
- C20 Rod Control and Information System (RC&IS) (DL2 / SC3)

A contact is closed in the selected switch position to provide “continuity” to or from the receiver. The other three (not selected) received switch positions have no continuity. From the point of view of the receiver there are always three positions with no continuity and one position with continuity. The Mode Switch positions sent to the C20 DL4a (SC2) and C20 DL2 (SC3) systems are received by TMR controllers that are used to validate the Mode Switch input signals.

The receiving system logic determines the resolution of “broken” switch positions. For example, the DL3 fail-safe systems trip for multiple switch positions or no switch positions. Fail-as-is systems generally alarm and stop their processes. The Automatic Power Regulator takes the plant out of automation and the RC&IS blocks control rod motion. The systems alarm for inconsistent switch positions.

#### 7.3.1 System C10 – Defense Line 3 / Safety Class 1

The three divisions of the C10 SC1 I&C systems are powered by three separate divisions of the R10 electrical system. Each of the three divisions uses an Uninterruptible Power Supply (UPS) each backed by its own division of 72-hour batteries. The UPS and the battery chargers can be



powered by off-site power or either of the standby diesel generators. The C10 SC1 I&C system actuation functions are designed as fail-safe if power to the C10 signal acquisition, trip determination, or actuation devices is lost, sensor failure and sensor communication is lost or voting communication is lost the reactor scram, RPV and containment isolations, and the ICS is automatically initiated to remove decay heat. Fail-Safe, as defined for the BWRX-300 design, is the ability of a system or component to move to a predetermined state when a failure occurs that ensures no harm comes to equipment, people, or the environment. This definition is more restrictive than defined for fail-safe in CNSC REGDOC-3.6 (Reference 7.3-1), which is defined as the ability of a system or component to move to a predetermined safe state when a failure occurs, and the most probable failure modes do not result in a reduction of safety.

Each C10 set of DCIS cabinets, each R10 DL3 UPS, battery chargers, and each R10 DL3 battery is located in a separate, fire barrier area of the Reactor Building. The instrument racks that mount the signal transmitters that provide the C10 measured trip, isolation and initiation signals are located in separate fire barrier rooms in the Reactor Building.

The C10 SC1 I&C system acquires the plant information from the sensors specified for systems performing DL3 functions. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment is specified as part of the detailed design process. The plant information acquired by C10 is displayed on C10 VDUs and the C20 integrated plant displays developed through the Human Factors Engineering (HFE) process.

#### **7.3.1.1 Systems Architecture**

The main functions of DL3 logic are the initiation of the scram, reactor and containment isolation, and ICS initiation functions. The necessary DL3 parameters are input to the C10 SC1 I&C system per division. The C10 SC1 I&C system (per division) provides signal conditioning and uninterruptible power for C10 sensors. The C10 signals are hardwired per division to the C10 cabinets, and the cabinet outputs are hardwired to their actuators. Remote data acquisition multiplexers are not required because the C10 SC1 I&C system and their signals are mostly located in the Reactor Building. The functional architecture for the C10 SC1 I&C system is shown in Figure 7.3-2.

The sensor trip logic is similar for each function:

- Each division measures a parameter using its own sensors
- Each division compares the measured parameter to a setpoint
- Each division determines if the measured parameter exceeds the setpoint

The three divisions communicate (i.e., vote) among themselves to determine if there have been sensor trips in 2-out-of-3 divisions. Each division independently makes the 2-out-of-3 trip decision. The three-division sensor and logic design is shown in Figure 7.3-3.

The data acquisition functions are implemented on C10 SC1 I&C hardware and software platforms in the C10 cabinets with a common backbone to the C10 communications function. The backbone ensures that C10 SC1 I&C platforms in a division have access to the C10 signals in that division and the communication function ensures that divisional data are isolated and sent through a unidirectional boundary device to the C20 gateways.

As an example, a reactor pressure signal may be used for both the C10 DL3 Hydraulic Scram and ICS initiation. Any C10 SC1 I&C actuation signal within a division is available to any trip logic. Any signal within a division is available for display and alarming (per division) on the appropriate

divisional VDUs. Any C10 SC1 I&C signal is available via an isolated link through unidirectional boundary device to the C20 systems.

The voting communication between divisions is limited to message authentication and divisional or parameter trip status, no other communication is included and if there is a loss of communication from a division, the receiving division assumes a trip.

The divisions have additional logic to support monitoring and include an alarm system for predetermined parameters (per division on the corresponding divisional VDU). This design supports accident monitoring for DL3 / Safety Category 1 signals and the corresponding divisional support systems like electrical supply (R10) status. The operator can input a limited set of commands (per division on the corresponding divisional VDU). The C10 divisions export plant process, accident monitoring, and diagnostic data via dedicated, unidirectional boundary devices to the C20 systems.

Reactor scram, reactor and containment isolation, and ICS initiation can also be initiated without software via hard switches in either plant control room.

### **7.3.1.2 System Design Bases and Associated Safety Functions**

The design basis of the C10 SC1 I&C systems is to mitigate the effects of a PIE (i.e., AOO or DBA and most DEC's) assuming no credit for DL2 (e.g., CCF). The main functions of the C10 SC1 I&C logic are the initiation of the following DL3 functions for reactor scram, reactor and containment isolation, and ICS initiation functions. The C10 SC1 I&C system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states and PIE in order to ensure that adequate information can be obtained on plant status. DL3 operating by itself ensures that no transient or design basis accident causes a radiation release greater than regulatory requirements. DL3 operates independently of DL2 using diverse equipment to perform required DL3 functions. The supporting safety analyses are described in Chapter 15.

C10 SC1 I&C system is designed to be available during all modes of plant operation. Specific safety functions actions are mode dependent, as determined by the Fault Evaluation process described in Chapter 15, Section 15.2, and specified by the Mode Switch position shown for each function in Table 7.3-1.

The DL3 functions categorized as Safety Category 1 and implemented in C10 with SC1 I&C equipment are:

1. Hydraulic Scram - Initiates a trip when a measured parameter exceeds a predefined setpoint. Initiates a hydraulic scram with a 2-out-of-3 voting logic using the same parameter. The trip logic is used to scram the reactor by hydraulically inserting the control rod blades. Hydraulic Control Units (HCU) are pressurized accumulators used to forcibly drive the control rod blade into the core. Scram initiation is enabled by de-energizing two scram solenoid valves to vent air holding the HCU scram valve closed against a spring force. When the HCU scram valve is opened, two control rod blades are inserted into the core except for one HCU that inserts a single control rod blade. The C10 DL3 Hydraulic Scram functions and initiation signals are listed in Table 7.3-1.

The HCUs are arranged in four spatial groups (3 with 7 HCUs and 1 with 8 HCUs). 29 HCUs are used to drive the 57 control blades. C10 uses a pair of divisional load drivers to de-energize the scram solenoid valves in each group of HCUs.

Simultaneous with the initiation of the DL3 Hydraulic Scram, a scram follow signal is sent to C20 DL4a to have each FMCRD motor drive its ball nut upward to a position just below the fully inserted and latched hollow piston that is coupled to the control rod blade. This signal is

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

a Non-Safety Class function. This signal is sent whenever a hydraulic scram is demanded by C10 SC1 I&C regardless of whether the hydraulic scram actuation has been successful.

More information on the Control Rod Drive System is found in Chapter 4, Subsection 4.5.1.

2. Power Range Neutron Monitoring (PRNM) - C10 SC1 I&C acquires and processes the PRNM neutron monitoring power signals to support the C10 DL3 Hydraulic Scram function. The PRNM consists of 13 vertical “strings” of Local Power Range Monitors (LPRM) arranged radially throughout the BWRX-300 core. Each string has four LPRMs equally spaced vertically and equally along the height of the core. The LPRMs signals are averaged per division to produce an APRM signal. The averaged signal is normalized to thermal power in percent. PRNM provides high power APRM flux trips to C10 DL3 Hydraulic Scram function when that parameter exceeds its setpoint.

PRNM also provides high simulated thermal power trips to C10 DL3 Hydraulic Scram function when that parameter exceeds its setpoint. The APRM signal is put through a (typical) six second time constant to relate it to the heat flux (as opposed to neutron flux) through the fuel to produce the STP signal (also in percent thermal power). PRNM provides high STP trips to C10 DL3 Hydraulic Scram function when that parameter exceeds its setpoint. Because there are transients where neutron flux can spike but thermal flux does not, the STP trip setpoints for C10 DL3 Hydraulic Scram function are set lower than the APRM (neutron) flux signal.

The PRNM also provides signals for accident monitoring purposes and to the 3D core thermal power distribution monitoring program and to the DL2 control rod blocking systems. These additional functions are not required to support any DL3 / Safety Category 1 function; however, C10 SC1 I&C is used to acquire the sensor inputs.

The individual LPRM signals are sent via isolated data links through unidirectional boundary devices to the 3D core thermal power distribution monitoring program. That program determines the information needed to calibrate the LPRMs.

More information on Core Monitoring is found in Chapter 4, Subsection 4.5.3.

3. Reactor, Containment, and System Isolation – C10 SC1 I&C performs the DL3 functions for RPV and containment isolation by closing the Main Steam Reactor Isolation Valves (MSRIV), Main Steam Containment Isolation Valves (MSCIV), Feedwater Reactor Isolation Valves (FWRIV), and Feedwater Containment Isolation Valves (FWCIV) and other valves listed in Chapter 6, Table 6.3.4-1 to limit line break effects both inside and outside containment. The isolation functions and initiation signals are listed in Table 7.3-1. The different isolation signals are initiated when 2-out-of-3 divisions agree on an isolation demand.

More information on the different isolation valves is found in Chapter 5, Section 5.10, and Chapter 6, Subsection 6.3.4.1.

4. Isolation Condenser System Actuation - C10 SC1 I&C performs the DL3 function to initiate ICS, which is comprised of three ICs. The three BWRX-300 ICs are each a simple loop including the vessel steam supply, piping to the radiators (i.e., heat exchangers) submerged in the ICS pools and a line to return the condensed steam to the vessel. Each IC steam supply line and condensate return line are equipped with two series normally open isolation valves mounted directly to the reactor vessel. The ICS is initiated by opening the normally closed condensate return. The C10 SC1 I&C logic activates each IC separately using 2-out-of-3 voting logic. When initiated by high reactor pressure, a different initiation setpoint is used for each IC. An IC is initiated by opening a normally closed condensate return valve per IC. One of the valve solenoids is in division 1 and the other valve solenoid is in division 2. Each solenoid has a normally closed load driver that opens on a 2-out-of-3 vote. Because the ICS

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

is necessary for the Safety Category 1 function, no lesser classified system can isolate them. Only the C10 SC1 I&C system can isolate an IC. The ICS functions and initiation signals are listed in Table 7.3-1.

More information on the ICS is found in Chapter 6, Section 6.2.

5. Diverse Isolation Condenser System Isolation - Because of the importance to BWRX-300 of off-normal heat removal, inventory maintenance, and pressurization limitation, the IC isolations use a fail-as-is design. The C10 SC1 I&C logic actively energizes the isolation valve solenoids to isolate an IC, and each IC is isolated independently and only upon indication of a line break for that IC. The IC isolation controls are placed into two groups. One group has the isolation controls for two ICs, and the second group has the isolation controls for remaining IC. For the first IC group, the normal DL3/SC1 hardware/software platforms perform the IC isolation function. For the second group (one IC), a separate and diverse DL3/SC1 hardware/software platform performs the isolation function. This arrangement ensures that a CCF affecting the normal DL3/SC1 hardware/software platform that causes a spurious isolation of ICs does not lead to a loss of ICS functionality. The diverse hardware/software platform utilizes redundant logic such that a single failure does not cause the isolation of the IC it controls.

More information on the Isolation Condenser System Isolation is found in Chapter 6, Subsection 6.2.7.2.

6. Monitoring – C10 SC1 I&C acquires accident monitoring information associated with Type A, B, and C accident monitoring variables, as defined in IEC 63147 (Reference 7.3-2). Type A variables provide the primary information required to permit the operating staff to take specific planned manually controlled actions for which no automatic control is provided and are required for safety systems to perform their protection functions as assumed in the plant anticipated operational occurrence and accident analysis licensing basis. The BWRX-300 does not have any Type A variables, as noted in Table 7.3-2. Type B are defined variables that provide primary information to the accident management personnel to assess the plant safety functions. These safety functions include accomplishing or maintaining reactivity control, core cooling, reactor coolant system integrity, and containment integrity (including radioactive effluent control). Type C variables provide primary information to the accident management personnel to indicate the potential for breach or the actual breach of fission product barriers (e.g., fuel cladding, reactor coolant system pressure boundary, and containment pressure boundary).
7. Communications - Used to acquire the process, diagnostic, and monitoring signals within the divisional C10 SC1 I&C components and organize them into a predefined message protocol. The divisional communication messaging is then isolated using SC1 hardware and converted into two optical fiber message streams per division. The three redundant pairs of divisional optical fiber message streams are sent to two SC2 gateways and then to the DL2 networks through unidirectional boundary devices. The gateways are redundant such that if one fails, the three divisions of C10 SC1 I&C signals are still available for alarming, recording, and monitoring in the lesser DLs.

The Secondary Control Room HVAC is normally in service and controlled by C20 DL2 equipment. The related C10 I&C logic is used for monitoring, isolation, and initiating filtered air circulation. The logic also monitors for SCR outside air radiation and automatically switch the normal air intakes to filtered outside air. These functions are not yet categorized. More information on the Control Room Habitability System is found in Chapter 6, Section 6.4.

The setpoints for C10 SC1 I&C actuation logic are determined with the setpoint methodology defined in IEC 61888 (Reference 7.3-3) using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the C10 SC1 I&C equipment (e.g., sensors and processing units). The design basis of DL3 is to act when DL2 is not effective at intercepting a transient or when an event is beyond the capabilities of the DL2 functions. In the event that DL3 completely fails, DL4a is designed to prevent core damage. As such, for common physical parameters measured by the DLs, the DL2 setpoint would be designed to act first, since its design basis is to mitigate/prevent DL3 from needing to respond. DL3 and DL4a would have common analytical limits to simplify transient and accident analyses and application of the formal setpoint methodology. The use of a common analytical basis has no adverse impact on plant operations or safety.

The BWRX-300 does not use interlocks to prevent overpressurization of low-pressure systems, interlocks to prevent overpressurization of the reactor coolant system during low temperature conditions, interlocks to isolate Safety Class systems from Non-Safety Class systems and interlocks to preclude inadvertent interconnections between redundant or diverse safety systems for the purposes of testing or maintenance. No interlocks are required because the BWRX-300 design eliminates to the extent practicable piping systems, major system components (e.g., pumps and valves), and subsystems connected to the reactor coolant pressure boundary to an ultimate rupture strength at least equal to the full reactor coolant pressure, as described in Chapter 5.

#### **7.3.1.3 Fundamental Design Properties in the System Design**

The C10 SC1 I&C system design includes DL1 properties that represent the quality measures implemented to minimize the potential for failures to occur by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and D-in-D features are discussed in the following five subsections.

##### **7.3.1.3.1 *Qualified***

The C10 SC1 I&C equipment is designed for to operate in the environment that would be expected during both normal operations and anticipated off-normal conditions. The C10 SC1 I&C equipment is qualified to perform its intended functions. Qualification addresses both hardware and software aspects of the C10 SC1 I&C system. The C10 SC1 I&C design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The C10 SC1 I&C system equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The C10 SC1 I&C qualification measures confirm that the I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The C10 SC1 I&C hardware platform used to perform its functions is developed in accordance with IEC 60987 (Reference 7.3-4) and meets IEC 61513 (Reference 7.3-5) requirements and the equipment standards defined in Section 7 in IEC 61226 2020 (Reference 7.3-6) for Class 1 systems. DL3 software used to perform its functions is developed in accordance with IEC 60880 Category A requirements (Reference 7.3-7). DL3 hardware used to perform its functions is qualified for environmental conditions in accordance with IEC 60780 (Reference 7.3-8) and seismic interaction in accordance with IEC 60980 (Reference 7.3-9). The C10 SC1 I&C equipment is qualified for electromagnetic compatibility in accordance with IEC 61000-4 (References 7.3-10 through 7.3-26) and IEC 61000-6-2 (Reference 7.3-27).

#### **7.3.1.3.2 Reliable**

The C10 SC1 I&C systems have the required reliability to perform its intended functions. It has an initial quantitative reliability target of less than 1E-4 probability of failure on demand. These reliability targets are achievable based on GEH experience with similar nuclear power plant protection systems. The reliability of the C10 SC1 I&C system is demonstrated as part of the final design.

The reliability analysis of the C10 SC1 I&C systems uses qualitative and quantitative performance measures or criteria, as appropriate, to demonstrate reliability goals. The reliability assessment is used to optimize goals such as minimizing out-of-service time for repair and reducing the frequency of surveillance testing.

The C10 SC1 I&C systems use digital technology and different C10 SC1 I&C functions may be combined on a common platform. It uses three divisions with 2-out-of-3 voting logic for trips and initiations. Spurious actuations due to single hardware failures are prevented.

The C10 SC1 I&C design allows one division of sensors to be bypassed or one division of logic to be bypassed to support maintenance or testing. The sensor and logic bypass switches are used to support these activities. The use of measuring and test equipment that can impair a C10 SC1 I&C function requires deliberate manual intervention via hardware interlock features at the system interfaces.

The individual LPRMs can be bypassed to allow the updated sensor calibrated values to be periodically input to the logic as required by plant Operational Limits and Conditions (OLC). APRM signals are periodically calibrated to current core thermal power. The LPRM averaging logic that is used to calculate the APRM value monitors the number of bypassed LPRMs going into the average. If the number drops below a predefined amount, the average is declared invalid, and the downstream logic assumes an APRM trip for that division. The LPRM signals contributing to the APRM and STP signals can be bypassed one at a time.

Correct functioning of the C10 SC1 digital I&C platforms are continuously monitored by self-testing features rather than by manual surveillance testing. Critical faults detected by the self-testing features trip the outputs of the platform and other faults are alarmed. The C10 system is designed to support required surveillance testing without affecting plant operation. It provides for signal and division bypass to support maintenance and testing. Bypass conditions are alarmed in the MCR. The C10 SC1 I&C design provide for periodic testing of the entire division of instrumentation logic, from sensing device to actuating device.

The C10 SC1 I&C platforms and diagnostics and their communication to the C20 Nuclear Segment are designed to support online and automatic surveillance testing that meets OLC requirements. Testing and performance assessment requirements are developed using the guidance of IEC 60671 (Reference 7.3-28) and IEC 62385 (Reference 7.3-29). Redundant divisions are monitored for deviations by an automated function that is performed continuously in the OLC Monitor described in Subsection 7.3.3. Deviations greater than a specified value are alarmed.

One of the scram solenoids uses a load driver and uninterruptible SC1 power from division 1 and the other load driver from SC1 power from division 2. The division 1 and division 2 load drivers open on a 2-out-of-3 vote from the three divisions communicated via optical fiber. Both scram solenoids per HCU are de-energized to initiate a scram, this design allows surveillance testing of each division through to the solenoid without causing a scram in normal operation. Testing provisions that are permanently connected to safety systems are classified the same as the safety system. The arrangement of the scram solenoid control and power is shown as Figure 7.3-4.

Although Figure 7.3-4 indicates a single load driver in division 1 and in division 2, in fact there are four load drivers in each. The four load drivers in each division are assigned to the scram solenoids in HCUs located in four separate rooms in the Reactor Building. In addition, the reactor can also be shut down with the complete failure of C10 DL3 Hydraulic Scram function with the diverse backup of DL4a. The four groups of cables from the load drivers to the four HCU rooms are in individual grounded conduits to preclude the possibility of a hot short keeping the solenoid incorrectly energized. Specifically, the circuit shown in Figure 7.3-4 is replicated four times per division 1 and division 2.

The Reactor and Containment isolation valves are closed by de-energizing two isolation valve solenoids mounted on the valve actuators. The valve motive power is typically stored nitrogen in accumulators independent of other nitrogen uses. One of the isolation solenoids uses a load driver and uninterruptible power from division 1 and the other solenoid is powered from division 2. The division 1 and division 2 load drivers open on a 2-out-of-3 vote from the three divisions communicated via optical fiber. Both isolation solenoids per isolation valve are de-energized to initiate a valve closure, this allows surveillance testing of each division through to the solenoid without causing an isolation in normal operation. The arrangement of the isolation solenoid control and power is shown as Figure 7.3-5.

The ICS actuation circuits have two normally energized solenoids on the IC open or close valve that keep it normally closed. One of the valve solenoids is in division 1 and the other valve solenoid is in division 2. Each solenoid has a normally closed load driver that opens on a 2-out-of-3 vote from the three divisions communicated via optical fiber. Both solenoids per actuation valve are de-energized to initiate a valve opening, this allows surveillance testing of each division through to the solenoid without causing ICS initiation in normal operation.

Each IC isolation valve is designed with three divisional solenoids, and 2-out-of-3 solenoids must change state for the valve to open or close. Reliable isolation and prevention of inadvertent isolation is achieved by the assignment of the C10 SC1 I&C divisions to solenoid valves that control each IC isolation valve and the mechanical arrangement of the solenoid valves to ensure that a single failure does not result in the unwanted isolation of an IC or prevent a required isolation of an IC. Manual actuation of the IC isolation function can be performed on a train-by-train basis. The manual isolation is a DL4a function, however it is implemented via SC1 equipment, because no lower classified equipment can have the ability to isolate an IC train (which would potentially prevent SC1 actuation of that train).

The C10 SC1 I&C system uses dual power supplies and dual power feeds per chassis to increase system reliability and availability.

#### **7.3.1.3.3 Robust**

Robustness of the C10 SC1 I&C design is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust C10 SC1 I&C design reflects the use of design methods and adherence to engineering best practices to ensure its functions are achieved for all operational states and accident conditions. The C10 SC1 I&C design requirements address the full range of operating environments associated with normal operation, transient, and accident conditions, as well as foreseeable internal and external hazards.

The C10 SC1 I&C systems has the required separation and independence to perform its intended functions.

The C10 SC1 I&C equipment is located in three separate divisional fire barrier rooms in the Reactor Building. The equipment is separated from the C20 equipment that is located in a separate fire barrier room in the Control Building.

C10 SC1 I&C system uses sensors and actuators independent of the C20 DPS.

The C10 SC1 I&C system provides trip communication between divisions for voting using optical fiber. A unidirectional boundary device provides for isolated optical communications to C20 systems through the C20 gateways to prevent communication from lower safety class systems from affecting C10 SC1 I&C systems. The communication protocols meet the data communication requirements of IEC 61500 for Category A functions (Reference 7.3-30). The C10 isolation devices are classified as SC1.

The logic used to rapidly scram the reactor by hydraulically inserting the control rod blades is separate and independent of the normal FMCRD positioning controls using the RC&IS

Each PRNM string has four LPRMs equally spaced vertically along the approximate 3.7 meter height of the core. The LPRM string is located between the corners of the fuel channels of the adjacent four fuel bundles. The locations are determined such that the 3D core thermal power distribution monitoring program can use mirror and rotational symmetry to virtually rotate and reflect the LPRM readings for unmonitored locations with similar surrounding fuel to gain complete core coverage. Figure 7.3-6 indicates the LPRM locations in the BWRX-300 core.

Each LPRM is individually powered and signal conditioned by its associated division and then averaged to generate APRM and STP values, which are compared to setpoints to determine if a scram is required.

The C10 SC1 I&C physical separation and signal isolation devices meet the requirements of IEC 60709 (Reference 7.3-31).

The C10 SC1 I&C systems are designed with three divisions with 2-out-of-3 voting logic for trips, isolations, and initiations. This redundancy ensures that the protection functions can be actuated even with an assumed single failure. It is only physically possible to bypass one division of sensors or one division of logic at a time, which retains the capability to provide required safety actuations. When a bypass is used, the 2-out-of-3 logic essentially operates as the equivalent of 2-out-of-2 to trip, isolate or initiate. Bypass conditions are alarmed in the MCR. The 3-fold redundancy scheme is acceptable based on high reliability of the DL3 system, administrative controls limit the time allowed for removal from service for maintenance bypass, and the availability of DL4a as D-in-D to ensure that the overall plant safety goals are satisfied.

The C10 SC1 I&C system has the required fail-safe design features to perform its intended functions. Deliberate operator action is required to return the C10 SC1 I&C system to normal after actuation.

The C10 DL3 Hydraulic Scram, PRNM, and Reactor and Containment isolation functions use only fail-safe logic (including loss of sensor and communications data). The C10 DL3 Hydraulic Scram logic is also equipped with individual manual scram switches in division 1 and division 2. These switches interrupt the power to the scram solenoids independently of the C10 DL3 Hydraulic Scram logic and using no software. The switches are included in the SCR and both switches are required to be operated to cause a scram. A loss of uninterruptible power or logic power causes a C10 DL3 Hydraulic Scram and RPV and containment isolation.

The DL3 / Safety Category 1 PRNM logic provides flux related trip signals to C10 DL3 Hydraulic Scram. It also provides signals to Safety Category 3 blocking functions (see Subsection 7.3.3) and to computer functions that determine the 3D power distribution of the core. A block diagram of the PRNM is shown as Figure 7.3-7.

The LPRM averaging logic that is used to calculate the APRM value monitors the number of available LPRMs going into the average to ensure it is representative. If the number drops below



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

a predefined amount, the average is declared invalid, and the downstream logic assumes an APRM trip for that division.

The APRM logic receives inputs from the Reactor Mode Switch position. With the Reactor Mode Switch in RUN, certain APRM trips are enabled and with the Reactor Mode Switch position in STARTUP enables different C10 DL3 trip functions.

The C10 ICS actuation logic initiates the three ICs separately (using 2-out-of-3 voting logic) for both reliability reasons and to allow different initiation setpoints for each IC. A loss of power initiates the ICs. The other condensate return valve is open/close only and is operated by DL3 / Safety Category 1 logic. This ICS actuation function is independent of the ICS throttling function. The logic is similar to the C10 DL3 Hydraulic Scram function logic.

Because of the importance of the ICS to BWRX-300 design to provide heat removal, inventory maintenance, and overpressurization protection, the ICs isolations do not use fail-safe design. Instead, the IC steam supply and IC condensate return isolation valves are fail-as-is to minimize the potential for inadvertent isolation. Additional design features are provided to ensure that the ICS provides the required safety functions. The C10 SC1 I&C logic actively energizes the IC isolation valve solenoids to isolate an IC and each IC is isolated independently and only upon indication of a line break for that IC.

Similarly, the isolation logic and power are fail-as-is. This IC isolation logic works by each division acquiring the isolation parameters and allows the divisional sensor to be bypassed one at a time. Bypass conditions are alarmed in the MCR. The signals are compared to their setpoints and initiate a trip (per division) if the sensed signal exceeds the setpoint. The three divisions communicate their trip decisions, and each division determines if at least two divisions have tripped to initiate an isolation. Unlike 'fail-safe' logic, which treats an invalid input or loss of communication as a vote to trip, fail as-is logic ignores invalid inputs such that the logic always requires two valid trip votes prior to actuation.

Then the load drivers compare their isolation trips from the three divisions, and if at least two divisions have tripped, an isolation is initiated. If logic or valve solenoid power is lost or the logic fails self-diagnostic monitoring, the isolation valves remain at their last position. From this point the fail-as-is logic differs from the fail-safe logic used for C10 SC1 I&C actuation features. The divisional logic outputs change state (open or close the isolation valves) only with active power. If logic or valve solenoid power is lost or the logic fails self-diagnostic monitoring, the isolation valves remain at their last position. Logic and sensor diagnostics are alarmed but do not initiate an isolation. A loss of power to the division or to the solenoids does not initiate an isolation. The isolation is active and latching to fail-as-is. To meet redundancy requirements, there are three solenoids on each isolation valve, and energizing any 2-out-of-3 solenoids cause the valve to change position and latch. Reliable isolation, when required, is achieved by requiring 2-out-of-3 solenoids and 2-out-of-3 divisional logics to initiate an isolation. The 2-out-of-3 logic minimizes the potential for inadvertent isolation for random I&C component failures.

The IC isolation controls are placed into two groups to prevent the remote possibility of a DL3 CCF causing three ICs to be spuriously isolated. One group has the isolation controls for two ICs and the second group has the isolation controls for remaining IC. For the first IC group, the main C10 hardware/software platforms perform the IC isolation function. For the second group (one IC) a separate and diverse hardware/software platform performs the isolation function. The diverse hardware/software platform is internally redundant such that a single failure does not cause the isolation of the IC it controls. The IC isolation function uses 2-out-of-3 logic to isolate an IC. The ICS design ensures that even with a postulated CCF of the primary C10 SC1 I&C platform, at least one IC is maintained operational, and one IC is sufficient to mitigate AOOs.

A critical self-diagnostic hardware or software failure initiates a divisional trip for fail-safe logic circuits. Diagnostic failures may be classified as critical (such as the watchdog timer or loss of communication) and cause a trip in that division. Other diagnostics are non-critical (e.g., one of the redundant power supplies fails) and cause an alarm in that division.

No C10 SC1 I&C function is dependent on whether the optical fiber unidirectional boundary devices are connected or disconnected, whether the fibers are severed, the gateways are powered, or the C20 networks have received the C10 signals.

Analog technology is used in C10 SC1 I&C systems for the bypass of one division of sensor inputs or one division of logic at a time. A software free interface is used in C10 to provide hardwired manual scram and RPV isolation capabilities.

#### **7.3.1.3.4    *Secure***

The C10 SC1 I&C systems have features that adequately addresses access control to limit cyber security vulnerabilities and ensure the system can perform its intended functions. The C10 SC1 I&C design incorporates access control features to support establishment of a secure operational environment. The C10 SC1 I&C software is produced in a secure development environment that prevents the insertion of undocumented codes. The C10 SC1 I&C systems are installed and maintained in accordance with the station administrative procedures and control of access programs.

The C10 SC1 I&C network is isolated from the rest of the BWRX-300 DCIS. The three SC1 DCIS divisions operate completely independently from the SC2 and SC3 DCIS. Information only flows through unidirectional boundary devices from C10 SC1 I&C to the SC3 networks. The C20 Nuclear Segment includes the SC2 gateways receive isolated SC1 signals through a unidirectional boundary device and provide them to the C20 Nuclear Segment. The SC2 gateways are discussed in Subsection 7.3.3. Measured parameters, setpoints, logic and trip status, and diagnostic information are continuously sent through these gateways to the C20 Nuclear Segment OLC Monitor, which can perform online surveillance testing, alarm on discrepancies, and additionally alarm on loss of these expected communications.

#### **7.3.1.3.5    *Defence-In-Depth***

The C10 SC1 I&C system provides the main line of protection for AOOs and DBAs. DL2 Nuclear Segment provides anticipatory trip functions to support the D-in-D concepts. DL2 is designed to prevent or mitigate AOOs before either DL3 or DL4a is required. The design basis of the DL4a systems provide D-in-D for the complete failure of the DL3 / SC1 systems (i.e., CCF). DL4a independently provides comparable DL3 plant protection functions using diverse platforms from DL3.

Diversity measures are incorporated within the C10 SC1 I&C system design as an additional level of protection for potential systematic faults caused by design and implementation defects within redundant divisions of the system. The IC isolation controls are placed into two groups to prevent the remote possibility of a DL3 CCF causing three ICs to be spuriously isolated. One group has the isolation controls for two ICs and the second group has the isolation controls for remaining IC. For the first IC group, the main C10 SC1 I&C hardware/software platforms perform the IC isolation function. For the second group (one IC) a separate and diverse hardware/software platform performs the isolation function. The diverse hardware/software platform is internally redundant such that a single failure does not cause the isolation of the IC it controls. These diversity measures ensure that at least one IC remains operational for any CCF affecting the IC isolation features and one IC is enough to mitigate AOOs.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

The C10 DL3 Hydraulic Scram function logic is equipped with individual manual scram switches in division 1 and division 2 (as shown on Figure 7.3-4). These switches use a software free method to interrupt the power to the scram solenoids independently of the automatic actuation logic and using no software. The switches are included in the SCR and both switches are required to be operated to cause a scram.

The C10 ICS initiation logic is equipped with individual switches in division 1 and division 2 to open the ICS condensate return line valves. These switches use a software free method to close the valves. The switches are included in the SCR and both switches required to be operated to cause a scram.

The C10 SC1 I&C reactor, containment, and system isolation logic is equipped with individual switches in division 1 and division 2 to close the MSRIV, MSCIV, FWRIV, and FWCIV. These switches use a software free method to close the valves. The switches are included in the SCR and both switches required to be operated to cause a scram.

#### **7.3.1.4 Operator Interface**

C10 SC1 I&C data are available on their appropriate divisional displays and these displays are subject to a HFE evaluation to maximize operator responsiveness. Each division has its own VDU in both the MCR and SCR and locally in the C10 SC1 I&C equipment rooms. Each divisional display is dedicated to monitoring or controlling equipment in its associated division. The C10 displays also include an alarm system to provide for operator awareness and to prompt to the displays containing further information relating to the alarm. C10 SC1 I&C system bypass conditions are alarmed in the MCR. The displays are menu driven and it is possible to switch between related displays without going through the main menu.

A reactor scram, ICS initiation, and reactor, containment, and system isolation can be manually initiated by SC2 switches in the MCR and SC1 switches in the SCR. Manual actuation of the IC isolation function can be performed on a train-by-train basis.

Some C10 SC1 I&C monitoring signals provide necessary information to achieve or verify correct plant response to transients and accidents. The necessary signals are determined through HFE analysis and applicable regulatory requirements and are referred to as accident monitoring signals.

The C10 SC1 I&C signals are alarmed to prompt the operator to an appropriate display (e.g., range check problems or alarm setpoints exceeded). The C10 SC1 I&C signals and the various C10 SC1 I&C components are alarmed for self-diagnostics.

The C10 SC1 I&C monitoring functions are supported for 72 hours by the R10 UPS and batteries with no off-site power available.

C10 SC1 I&C data is also transmitted through an optically isolated data link through a unidirectional boundary device to C20, where the signals are recorded, alarmed, and displayed. The C10 monitoring also includes signals needed to monitor divisional support equipment like the UPS and battery chargers as well as the SC1 equipment room temperatures. The operator can view the C10 information on the C10 VDUs even if C20 is not operational.

The C10 SC1 I&C system acquires and displays the accident monitoring information associated with Type A, B, and C accident monitoring variables.

The BWRX-300 plant-level safety D-in-D design has a direct bearing on the design of the accident monitoring instrumentation. The BWRX-300 passive reactor design automatically actuates DL3 systems that rely solely on natural passive mechanisms based on fundamental physical and thermodynamic principles to mitigate a design basis event (i.e., anticipated operational

occurrences or postulated accident). The BWRX-300 does not require manual actions to mitigate design basis events. Automatic controls perform mitigation functions. As such, no Type A accident monitoring instruments are required. The simplicity of the BWRX-300 safety features to mitigate design basis events and safely shut the plant down, when coupled with the D-in-D framework, ensures the capability to mitigate the effects of design basis transients and accidents even with a coincident loss of a complete DL.

The simplicity and D-in-D framework change the role of the Type B and C accident monitoring instruments for the BWRX-300 accident management strategy. For the BWRX-300, these instruments are only used for immediate verification that DL protection functions have actuated. They are not needed for subsequent operator actions to realign or control more complicated engineered safety features after the initial actuation phase. The multiple DLs and their capability to accommodate the loss of a complete defense line eliminate the need for more immediate operator actions to respond to equipment failures for successful accident management. The specific accident monitoring variables that the operator should monitor to ensure safety during an accident and the subsequent long-term stable shutdown phase are determined through the normal BWRX-300 design process using the human factors engineering process. The Type A (if later required), B, and C accident monitoring information is displayed in both the MCR located in the Control Building and the SCR located in the Reactor Building in the current concept design. The C10 accident monitoring information is sent through qualified isolation devices to I&C system C20, which is SC3.

A preliminary list of the Type B and C accident monitoring variable is provided in Table 7.3-2.

The DL3 operator interface HSI for C10 is described in Sections 7.5 and 7.6 for the MCR and SCR, respectively.

#### **7.3.1.5 Compliance Alignment**

Table 7.3-5 shows how the C10 SC1 I&C system aligns to meet the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603 (Reference 7.3-32).

### **7.3.2 System C20 – Defense Line 4a / Safety Class 2**

Section 7.2 states C20 performs functions that are part of two DLs and is divided into two Safety Classes accordingly (See Subsections 7.3.3 and 7.3.4). These two DLs complement the C10 SC1 I&C systems to provide a high availability for the plant protection functions.

The C20 system acquires the plant information from the sensors specified for systems performing DL4a functions. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment is specified as part of the detailed design process. The plant information acquired is displayed on the C20 integrated plant displays developed through the HFE process.

#### **7.3.2.1 Systems Architecture**

The functional architecture for the DL4a system is shown in Figure 7.3-8. The splitter architecture used for shared sensors is shown in Figure 7.3-9. A representative DPS block diagram for a reactor scram is shown in Figure 7.3-10 and should be viewed with Figure 7.3-4, which shows the scram solenoid power configuration.

Protection function signals within the DL4a are point-to-point wire or optical fiber, they do not use the plant networks. The RC&IS communicates with the FMCRD motor controllers (which perform a DL4a function) on a dedicated redundant network. The FMCRD network is not part of the plant

DCIS network and does not go through the network managed switches. This communication is not required for any DL4a protection function nor could interfere with any DL4a safety function.

### **7.3.2.2 System Design Bases and Associated Safety Functions**

The design bases of the DL4a systems assume the complete failure of the DL3 / SC1 systems (i.e., CCF). DL4a independently provides comparable DL3 plant protection functions using diverse platforms from DL3. The C20 DL4a system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states and PIE in order to ensure adequate information can be obtained on plant status. The supporting safety analyses are described in Chapter 15.

C20 Defense Line 4a is designed to be in service during all modes of plant operation. Specific safety functions actions are mode dependent, as determined by the Fault Evaluation process in Chapter 15, Section 15.2, and the initiating signals are enabled based on the Reactor Mode Switch position shown for each function in Table 7.3-3.

The system functions performed by DL4a are:

1. Diverse Protection System - Provides the logic for the DL4a protection functions. The DPS is analog and segmented into three channels. Input signals are measured, sent to comparison logic, and if the signals exceed their setpoints, produce trip or initiation signals per channel. The channel trip and initiation signals are then voted 2-out-of-3 to produce a DPS initiation or trip and to avoid single-failure inadvertent trips. The DPS functions include:
  - a. A hydraulic scram independent of DL3
  - b. RPV, containment, and system isolations independent of DL3
  - c. ICS initiation independent of DL3
  - d. A FMCRD motor fast run-in
  - e. Automatic Rod Insertion (ARI) pilot valve actuation

The hydraulic scram and ARI signals are initiated whenever a DL4a scram is demanded by DL4a regardless of whether the fast motor run-in actuation has been successful. The DL4a safety function for the motor driven fast run-in occurs on any hydraulic scram signal plus a signal on reactor power level indicating that the hydraulic scram was not successful.

The DPS functions and initiation signals are listed in Table 7.3-3. The DPS provides for manual initiations using switches located in the MCR. These are separate from the switches associated with C10 SC1 I&C which are located in the SCR.

2. Gamma Thermometers - Provide electrical signals from in core devices based on local gamma flux which is representative of core thermal power. It represents a completely diverse technology to the neutron detecting LPRMs used in DL3 PRNM system. There are two types of Gamma Thermometers (GT) "strings" located in the core. The first design locates 8 GTs in each of the 13 instrument tubes containing LPRM detectors. The second design locates 8 GTs in each of the 10 instrument tubes containing WRNM detectors. The GT design uses a total of 184 GTs measuring dispersed radial and axial local power in the core. The net result is that both DL2 and DL4a DPS receive four analog signals representing core power using only analog components that are completely diverse from the DL3 LPRMs and APRMs. The GT signals are used as inputs to DPS.
3. Fine Motion Control Rod Drive Motor Control - There are 57 FMCRDs in the BWRX-300 and each one has an individual motor and an individual motor controller. The motor controllers are allocated to DL4a and interface with DPS, Emergency Rod Insertion Panels (ERIP), and

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

RC&IS. Each motor controller uses software to supply the appropriate commands to the FMCRD motors using individual position controllers (i.e., position demand and position feedback). Control rod position indication is provided to RC&IS. Normally the FMCRDs are individually (or in gangs) positioned by the RC&IS to control reactor power in response to either operator or automation system insert and withdraw commands. The motor controller software is required for the motor run-in, but the DPS actuated hydraulic scram does not use software.

4. Emergency Rod Insertion Panels and Emergency Rod Insertion Control Panels - There are four DL4a ERIPs. The ERIPs are relay-based and have two functions. The first is to multiply the ATS, C10 DL3 Hydraulic Scram, and DPS motor run-in demand signals to each of the 57 FMCRD motor controllers. The second is to multiply the Multi-Channel Rod Block Monitor (MRBM) rod block demand signals to each of the 57 FMCRD motor controllers. The MRBM rod block stops the rods from being able to withdraw and in the relay logic, the motor run-in has priority. The circuit design ensures control rod insertion is always available. The scram follow motor run-in is automatically initiated whenever ATS, DPS, or C10 DL3 Hydraulic Scram function initiates a hydraulic scram.

These panels are contact multipliers that each receive manual scram, three rod motor withdraw power block signals and three motor run-in signals from the Emergency Rod Insertion Control Panel (ERICP). The three motor run-in signals and three motor withdraw power block signals are decoded using 2-out-of-3 relay logic to pick up additional relays, 29 in one panel and 28 in the other panel. The design ensures that a relay failure can only affect one FMCRD. The motor run-in can be manually initiated by SC2 switches in the MCR initiate both a hydraulic scram and motor run-in. The motor run-in relays are sealed in once energized to ensure the motors have time to drive the rods full in.

The ERICP sends the manual scram and motor run-in and motor power block signals to the ERIP. The automatic motor run-in signals come from three sources. Two are, respectively, a hydraulic scram initiation signal from DL3/SC1 C10 DL3 Hydraulic Scram function and a hydraulic scram initiation signal from DL2/SC3 (ATS). Both hydraulic scram initiation signals use TMR controllers and point-to-point connections. A third motor run-in signal comes from DL4a/SC2 DPS that is also TMR.

The first two (DL3 Hydraulic Scram and ATS) hydraulic scram initiation signals are shared with DL4a to generate an FMCRD scram follow motor run-in. Simultaneous with the initiation of the DL3 and ATS hydraulic scrams, the scram follow motor run-in provides the additional post scram safety function of automatically driving each FMCRD motor drive its ball nut upward to a position just below the fully inserted and latched hollow piston that is coupled to the control rod blade. This, in addition to the SC1 internal control rod blade full in rod latches, prevents the control rod blade from dropping out of the reactor post scram. This also ensures that any rod scram failure (due to a failed HCU component, for instance) is inserted into the reactor within minutes of the scram signal. The DPS fast motor run-in is completely independent of the DL3 and DL2 scrams and serves as the backup to the complete failure of the hydraulic scram in case of complete failure of the mechanical/hydraulic portions of the Control Rod Drive (CRD) system. This function is performed by the ERICP sending signals to DPS to indicate that a hydraulic scram has been initiated from any source. After receiving such signals, the DPS logic expects its GT inputs to show a decreasing power due to a hydraulic scram. If the GT measurements do not decrease sufficiently within a short time, the DPS generates outputs to the ERICP to order a fast motor run-in. The FMCRD motors should have already started in their scram follow capacity and the DPS order to the ERICP causes the motors to increase insertion speed.

A motor power withdraw block signal comes from the Multi-Channel Rod Block Monitor (MRBM), which stops the rods from being able to withdraw. The circuit design ensures that control rod insertion is always available.

5. Fine Motion Control Rod Drive Motor Uninterruptible Power Supply - These UPS are dedicated to the FMCRD motor controllers and motors and not part of the R10 or R20 normal DCIS uninterruptible power supplies. The FMCRD motors and controllers have four dedicated UPS that can supply the FMCRD motors in the absence of off-site power or the absence of standby diesel generator power. These power sources are dedicated to the FMCRD, not shared with the SC2 R20 UPS that power the DL4a and DL2 DCIS.

The FMCRDs are divided into four groups distributed throughout the core. Each of the four motor and motor controller groups has its own UPS that operates the FMCRDs long enough to fully insert even if both off-site power and standby diesel generator power are lost. The four motor groups of FMCRDs are indicated by color code on the upper right of the core map in Figure 7.3-6.

The setpoints for C20 DPS actuation logic are determined with the setpoint methodology defined in IEC 61888 using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the DPS equipment (e.g., sensors and processing units). DL4a is designed to prevent core damage for PIE assuming the complete failure of DL3. As such, for common physical parameters measured by DL3 and DL4a would have common analytical limits to simplify transient and accident analyses and application of the formal setpoint methodology. The use of a common analytical basis has no adverse impact on plant operations or safety.

### **7.3.2.3 Fundamental Design Properties in the System Design**

The C20 Defense Line 4a SC2 system includes properties that represent the quality measures implemented to minimize potential for failures to occur by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and D-in-D features are discussed in the following five subsections.

#### **7.3.2.3.1 *Qualified***

The C20 SC2 equipment is designed for to operate in the environment that would be expected during both normal operations and anticipated off-normal conditions. The C20 Defense Line 4a equipment is qualified to perform its intended functions. Qualification addresses both hardware and software aspects of the C20 Defense Line 4a equipment. The design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The C20 Defense Line 4a equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The C20 Defense Line 4a equipment qualification measures confirm the I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The main C20 DL4a hardware platform is analog and meets IEC 61513 requirements and the equipment standards defined in Section 7 in IEC 61226 2020 requirements for Class 2 systems. DL4a software is used in the FMCRD motor controllers and is developed in accordance with IEC 62138 Category B requirements (Reference 7.3-33). DL4a hardware is developed in accordance with IEC 60987. DL4a hardware is qualified for environmental conditions in accordance with IEC 60780 and seismic interaction in accordance with IEC 60980. The C20 Defense Line 4a SC2 system equipment is qualified for electromagnetic compatibility in accordance with IEC 61000-4 and IEC 61000-6-2.

#### **7.3.2.3.2 Reliable**

The C20 Defense Line 4a equipment has the required reliability to perform its intended functions. It has an initial quantitative reliability target of 1E-3 to 1E-4, probability of failure on demand. The reliability analysis of the C20 Defense Line 4a equipment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate. The reliability assessment is used to optimize goals such as minimizing out-of-service time for repair and reducing the frequency of surveillance.

The DPS provides self-diagnostics to C20 for equipment status monitoring and use by RC&IS through qualified isolation features classified as SC2. The DPS is designed to support required surveillance testing without affecting plant operation. It provides for signal and channel bypass to support maintenance and testing. Bypass conditions are alarmed in the MCR. The C20 DL4a design provides for periodic testing of the entire channel of instrumentation logic, from sensing device to actuating device. Testing and performance assessment requirements for DPS are developed using the guidance of IEC 60671 and IEC 62385. The use of measuring and test equipment that can impair a C20 DL4a function requires deliberate manual intervention via hardware interlock features at the system interfaces. Testing provisions permanently connected to safety systems are classified the same as the safety system.

The GT system supports averaging the four groups of GT signals and normalizing the average to core thermal power. The GT system provides for bypassing individual GTs. Bypass conditions are alarmed in the MCR. The GT system provides for the calibration of the GTs. The GT system is designed to support required surveillance testing without affecting plant operation. Provides for GT system self-diagnostics for C20 display.

The FMCRD motor controllers provide self-diagnostics to DL2 C20 for monitoring and use by RC&IS.

The DPS, GTs and ERIPs in DL4a are redundantly and uninterruptedly powered by the two -out-of-three R20 electrical system load groups. The FMCRD motor controllers are normally powered from R20 and have dedicated UPSs. The FMCRD UPS have a capacity that is at least twice as long as the duration of a motor run-in (i.e., rods driven full in by their motors). The DPS, GTs and ERIP UPS are backed by two load groups of 72-hour batteries. The UPS and the battery chargers can be powered by off-site power or either of the standby diesel generators.

#### **7.3.2.3.3 Robust**

Robustness of C20 Defense Line 4a is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust C20 Defense Line 4a DPS design reflects the use of design methods and adherence to engineering best practices to ensure that the protection functions are achieved for the specified conditions.

The C20 Defense Line 4a systems have the required separation and independence to perform their intended functions as a diverse backup to DL3.

The majority of the C20 DL4a equipment is located in a separate fire barrier room in the Control Building with the remainder located in compartmentalized fire barrier rooms in the Reactor Building. It is separated from DL3 equipment that is located in three separate divisional fire barrier rooms in the Reactor Building. The DL4a equipment is powered by separate power supplies from DL3. The four ERIPs are located in the Reactor Building on opposite sides of the reactor and near the cabinets housing the FMCRD motor controllers.

The DPS uses sensors and actuators that are independent of DL3. Isolated communication is allowed from DL3 to DL4a but not vice versa. This communication cannot prevent either DL from fulfilling its protection functions. DL4a systems do not share any sensors with DL3.



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

DL4a is independent from DL2 to an extent that is practicable in that shared sensors are not credited with mitigation for the same PIE. Some signals shared between DL4a and DL2, as identified in Figure 7.3-8, are used by both DPS and DL2 TMR controllers allocated to each parameter. The shared signals and splitters are located in separate cabinets. The shared signals are developed using TMR architectures. The shared signals are analog signals. Signal sharing is accomplished using non-software, analog splitters and relays that are powered with redundant uninterruptable power supplies. The splitter technology supplies isolated outputs that allow short circuits or common mode voltages up to 300 volts (AC or DC) applied to one output without affecting the other output. The splitter design is shown in Figure 7.3-9. The splitter design ensures a splitter output is unable to adversely affect the other outputs. The C20 isolation features are classified as SC2.

The three channels of the DL4a DPS from the separate DL4a splitters are used by the 2-out-of-3 voting logic for trips and initiations.

The FMCRD motor controllers receive normal control signals from RC&IS in DL2; however, the motor controller SC2 software instructs the controller to ignore the SC3 RC&IS network commands and run the rods in when instructed by the relay-based ERIPs and analog DPS.

The control rods and motors are located inside containment in the Reactor Building. The FMCRD motor controllers are organized into four groups located in four separate rooms in the Reactor Building outside containment.

The DL4a FMCRD UPS provides the necessary independent power to the control rods and monitoring. The FMCRD motors and controllers have four dedicated UPS that can supply the FMCRD motors in the absence of off-site power or the absence of standby diesel generator power. The FMCRD UPS can operate from and be charged by those sources. The power sources are also independent of the normal SC2 R20 UPS that power the DL4a and DL2 DCIS. The four independent and separately located groups of ERIPs, FMCRD motor controllers and the UPS to power them are arranged such that the insertion of any three of the four groups will shut down the reactor.

The DL4a physical separation and signal isolation devices meet the requirements of IEC 60709.

Where software is used, DL4a diagnostics always include watchdog timers, sensor range checks, power supply monitoring, communications, and actuator monitoring. The DPS is analog but is monitored digitally through unidirectional isolated signals. The digital monitoring for self-diagnostics cannot adversely affect DL4a functions. The digital monitoring checks for inconsistencies between redundant signals, performs sensor range checks, and monitors actuator, communication, and power supply status. The relay logic is designed to ensure that a relay failure can only affect one FMCRD. The manual motor run-in can be initiated by SC2 switches in the MCR that initiate both a hydraulic scram and motor run-in. The motor run-in relays are sealed in for at least four minutes once energized to ensure the motors have time to drive the rods full in.

The C20 Defense Line 4a DPS has the required fail-as-is design features to perform its intended functions and avoid spurious actuations. The fail-as-is energize to actuate design is used to prevent lesser classified DPS from creating unnecessary challenges to plant safety (e.g., spurious actuations for expected failures) requiring action by DL3.

Deliberate operator action is required to return the C20 Defense Line 4a Protection System to normal after actuation.

DL4a operating by itself and assuming a complete failure of DL3, ensures that no transient or design basis accident causes a radiation release greater than regulatory requirements. No DL3

equipment can prevent a scram, reactor, and containment isolation, or ICS initiation from DL4a equipment or vice versa.

The C20 Defense Line 4a systems provide diversity to support the D-in-D concepts.

DPS initiates hydraulic scrams, isolations and ICS using the same solenoids as DL3 but using relays instead of load drivers. Independence between these DLs is discussed below.

For diversity, the DL4a is able to independently shutdown the plant using the FMCRD motors. The DL4a ERIPs accept isolated hydraulic scram demand signals from ATS and C10 DL3 Hydraulic Scram to also initiate reactor shutdown using the FMCRD motors. The motor run-in inserts rods even if the hydraulic scram does not work or the entire DL3 systems fail.

GTs provide a diverse technology to the neutron detecting LPRMs used in DL3 PRNM system. The GT system segment supplies two types of outputs: analog average power signals and digital signals of individual sensor measurements. The signals from the 184 GTs are sent to four GT data acquisition cabinets located in four separate rooms of the Reactor Building. The signals are apportioned to the four cabinets such that each cabinet gets a similar allocation of radial and axial GT signals. The signals are conditioned using analog equipment and then the signal paths split. The four analog signals are processed by an averaging circuit with adjustable gain. The averaging circuit averages the un-bypassed GTs and allows the gain to be periodically adjusted/normalized to core thermal power. The output of each cabinet is an analog 4 – 20 mA signal calibrated to 0 – 120% thermal power. DL4a DPS receive this signal representing core power using only analog components that are completely diverse from the DL3 LPRMs and APRMs. The digital GT signals are sent to DL2 via the C20 network.

The ERICP is relay only and not subject to common cause software failure.

#### **7.3.2.3.4    *Secure***

The C20 Defense Line 4a systems adequately address access control to limit cyber security vulnerabilities and ensure the system can perform its intended functions.

The C20 Defense Line 4a DPS uses on analog technology, which eliminates the major cyber security vulnerability for DL4a. The C20 Defense Line 4a FMCRD design incorporates features to support establishment of a secure operational environment. DL4a input signals and protection logic are analog and do not use digital technology, but the isolated diagnostics and internal data are available to the SC3 networks and controllers. The C20 FMCRD software is produced in a secure development environment that prevents the insertion of undocumented codes and precludes their use. The C20 Defense Line 4a digital equipment is installed and maintained in accordance with the station administrative procedures and control of access programs.

#### **7.3.2.3.5    *Defence-In-Depth***

The C20 Defense Line 4a systems provide diverse protection functions to place and maintain the plant in a safe state in the event of PIE concurrent with failure of DL3 functions to support the D-in-D concepts.

The design basis of the DL4a systems provide D-in-D for the complete failure of the DL3 / SC1 systems (i.e., CCF). DL4a independently provides comparable DL3 plant protection functions using diverse platforms from DL3. The DL4a sensors are selected to be as diverse as practical from DL3 sensors. The ERIP provides D-in-D by allowing DPS, C10 DL3 Hydraulic Scram, and ATS to drive in rods by the FMCRDs.

Diversity measures are incorporated within the C20 Defense Line 4a system designs as an additional level of protection for potential systematic faults caused by design and implementation defects within redundant divisions of the C10 SC1 I&C system. The C20 Defense Line 4a method

of actuation is diverse from the method of actuation implemented by C10 DL3 Hydraulic Scram function. The C20 Defense Line 4a actuation relays shown in Figures 7.3-4 and 7.3-5 are configured as energize to actuate whereas C10 SC1 I&C load drivers are configured as deenergize to actuate which provides functional diversity in these actuation circuits. The ARI pilot valves described in Chapter 4, Subsection 4.5.1 are energized to actuate and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods.

The FMCRD motor controllers are required to be diverse from the DL3 I&C platform equipment only if the DL4a motor run-in function is credited as a required mitigating function for a failure of C10 DL3 Hydraulic Scram function due to CCF in the SC1 I&C platform equipment.

The motor run-in can be manually initiated in the MCR (SC2) and SCR (SC1) with switches going to the ERIPs with no involvement of RC&IS software.

The FMCRD UPSs provide D-in-D if normal power is lost.

#### **7.3.2.4 Operator Interface**

The C20 Defense Line 4a diagnostic information is sent to other digital systems in DL2 to provide recording, monitoring, and alarming functions. C20 DL4a system bypass conditions are alarmed in the MCR. A reactor scram can be manually initiated by SC2 switches in the MCR and SC1 switches in the SCR.

The DL4a operator interface HSI for C20 is described in Sections 7.5 and 7.6 for the MCR and SCR, respectively.

#### **7.3.2.5 Compliance Alignment**

Table 7.3-6 shows how the C20 Defense Line 4a systems align to meet the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603.

### **7.3.3 System C20 – Defense Line 2 / Safety Class 3 / Nuclear Segment**

Section 7.2 states the C20 network is split into two segments: Nuclear and BOP. This section addresses the C20 Defense Line 2 Nuclear Segment, which complement the DL3 C10 SC1 I&C and DL4a C20 systems to provide a high availability for the plant protection functions. It also provides detail for the C20 Nuclear Segment which contains the DL2 ATS, major reactor control systems and the control rod blocking systems.

The C20 system acquires the plant information from the sensors specified for systems performing DL2 functions. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment is specified as part of the detailed design process. The plant information acquired is displayed on the C20 integrated plant displays developed through the HFE process.

#### **7.3.3.1 Systems Architecture**

The functional architecture for the C20 Nuclear Segment is shown in Figure 7.3-11. Depending on final DCIS design and processor loading, multiple functions may be combined and implemented on a common controller. A representative ATS block diagram for a reactor scram is shown in Figure 7.3-12. This figure should also be viewed with Figure 7.3-4, which shows the scram solenoid power configuration.

The C20 Nuclear Segment employ TMR and redundant controller architectures for processing units to improve reliability or eliminate spurious control actions caused by a single hardware failure, as shown in Figure 7.3.11.

Protection function and control signals needed between controllers for their functions are point-to-point wire or optical fiber.

The design provides both control and monitoring functions and, supports both local and remote data acquisition.

#### **7.3.3.2 System Design Bases and Associated Safety Functions**

The design basis of DL2 is to prevent or mitigate AOOs before either DL3 or DL4a is required. DL2 action precludes the need for other DL intervention. DL2 is classified as SC3 and the DL2 hardware/software platforms are diverse from DL3. The C20 DL2 system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states and during a PIE in order to ensure that plant status information can be obtained. The supporting safety analyses are described in Chapter 15.

C20 Defense Line 2 is designed to be in service during all modes of plant operation. Specific safety functions actions are mode dependent, as determined by the Fault Evaluation process in Chapter 15, Section 15.2, and specified by the Mode Switch position shown for each function in Table 7.3-4.

The main C20 Nuclear Segment functions are:

1. Anticipatory Trip System - The ATS is designed to perform the protection function of DL2 which is mitigating transients and providing investment protection for expected transients by tripping the plant in advance of any required DL3 or DL4a response. There is no failure of ATS that can adversely affect the protection functions of DL3 or DL4a. ATS uses logic similar to the C10 DL3 Hydraulic Scram function and DPS, in that parameters are measured, compared to a setpoint, voted and an actuator initiated. In the unlikely event the DL2 ATS fails to scram, the other independent and diverse DLs will still scram the reactor on flux or pressure to maintain the plant safety functions (to prevent unacceptable radiation release to the public). An ATS malfunction cannot stop a C10 Hydraulic Scram nor can a C10 DL3 Hydraulic Scram stop a DL2 ATS scram. The ATS scram initiation signals are listed in Table 7.3-4. Simultaneous with the initiation of the ATS hydraulic scram demand, a scram follow signal is generated such that each FMCRD motor drives its ball nut upward to a position just below the fully inserted and latched hollow piston that is coupled to the control rod blade. This signal is sent whenever a hydraulic scram is demanded by ATS regardless of whether the hydraulic scram actuation has been successful.

The C20 Nuclear Segment also provides DL2 non-scram functions, including:

- a. Feedwater runback for various transients
- b. Turbine Trip on Low Main Condenser Vacuum
- c. Turbine Bypass Valves Closure on Low Main Condenser Vacuum
- d. Turbine Bypass Valves Fast Open on Fast Closure of Turbine Control Valve / Turbine Stop Valve Demand
- e. MSRIV/MSCIV Isolation on Low Main Condenser Vacuum
- f. Standby Diesel Generator Start on Low Electrical Bus Voltage
- g. Feedwater and Condensate Pumps Trip on High RPV Level
- h. Feedwater Isolation on High-High RPV Level
- i. Start Standby Feedwater Pump on Loss of Operating Feedwater Pump

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

- j. Fuel Handling Area Ventilation Isolation on Detection of High Refuel Floor Radiation
  - k. Control Room Ventilation Isolation on Detection of High Refuel Floor Radiation
  - l. ICS Pressure Control on High Reactor Pressure
  - m. Maintain Target Reactor Pressure
  - n. Maintain Target Reactor Water Level
  - o. Control Rod Movement blocks
  - p. Select Control Rod Run-In (SCRRI) on Feedwater Temperature Decrease
  - q. Manual Shutdown Cooling System (SDC) Train Initiation
2. Plant Automation System - The Plant Automation System coordinates automatic reactor power control as part of a larger automation scheme, including coordination of C20 control systems (for control of nuclear equipment) and coordination of C30 control systems (for control of BOP equipment). The C20 and C30 control systems must individually be put into automatic mode by the operator for this automation to be functional; without this operator permissive, any commands originating from the Plant Automation System PAS are ignored.

The Plant Automation System provides batch control of the steps needed to automate the plant systems used to bring the reactor from cold conditions to rated power. One of these controllers is Automatic Power Regulator which interfaces with the nuclear control systems. Examples of these Automatic Power Regulator functions include:

- a. Bring reactor to critical
- b. Heatup and pressurization
- c. Increase power to 15%
- d. Roll turbine to rated speed
- e. Synchronize turbine-generator
- f. Increase electrical load to ~ 20%
- g. Automatic Power Regulator switch to power control

The Plant Automation System does not communicate with DL3/SC1 or the DL4a/SC2 I&C systems nor does it control SC1 plant equipment; instead, the operator is prompted to operate the Reactor Mode Switch to provide a necessary permissive before an automated function can occur.

After the minimum 20% power is reached by predefined automation steps, the Automatic Power Regulator accepts reactor power demands from approximately 20% to 100%. The power demand can be % thermal power or desired generator MWe as selected by the operator. If the regulatory body so allows, the desired generator MWe can come from a remote load dispatcher. In this case the operator sets an allowed maximum and minimum reactor power and the allowed rate of change in reactor power that supervises and limits load dispatcher inputs.

3. Reactor Level Control - Provides reactor level monitoring and control from cold conditions through pressurization and heatup to power operation. This system controls reactor level at greater than 10 – 15% power by adjusting the frequency of the adjustable speed drives supplying power to the feedwater pump motors. At lower powers the system uses a low flow control valve supplied with either a condensate pump or feedpump operating at fixed speeds,

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

depending upon reactor pressure. The Reactor Level Control (RLC) also operates level control valves in the shutdown cooling system and the Reactor Water Cleanup System when water is rejected from the reactor vessel during cold and heatup operations. The RLC generates a SCRR demand to RC&IS to obtain a rapid reactor power reduction if required by a feedwater temperature decrease.

4. Feedwater Temperature Control – Provides control of the feedwater temperature delivered to the reactor by the sixth stage feedwater heater by measuring the heater exit temperature and modulating a steam heating valve as required to meet the demanded temperature setpoint. The feedwater temperature is programmed as a function of reactor power and the temperature is held constant between 90 and 100% thermal power and then decreased as power is further reduced. Appropriate function logic features are provided to keep the feedwater nozzle temperatures from changing too rapidly. The controller also provides the sixth stage feedwater heater level control necessary to keep the feedwater heater level at the optimum setpoint for best heat transfer.
5. Reactor Pressure Control - Provides control of reactor pressure from a cold vessel through pressurization and heatup to power operation. During plant operation this system normally controls reactor pressure by sending a flow demand signal to the turbine controller. During plant startup or at other times when the turbine is off-line, Reactor Pressure Control (RPC) adjusts reactor pressure by sending a flow demand signal to the turbine bypass valves.
6. Isolation Condenser Throttling Pressure Control - To meet their safety requirements, the three BWRX-300 ICs have a very large capacity for reactor pressure reduction. While desirable in safety systems, this capacity can result in exceeding the normal allowed cooldown rate for the reactor vessel. The ICs are each initiated with two condensate return valves. One of these valves is “open or close” and can be actuated by DL3 only. The other valve is modulating and controlled by the IC throttling valve controller in DL2 but can be forced open by a DL4a initiation signal. The IC throttling valves can control IC flow by modulating their position. The throttling valves are controlled by a standard pressure controller and have staggered setpoints so that they do not initiate simultaneously but only as the reactor pressure increases. The IC throttling valve control is designed to act before ICS actuation by other DLs.
7. Turbine-Generator Control - The normal function of Turbine-Generator Control is to position the turbine control valves and bypass valves to control reactor pressure. The turbine-generator controller provides the turbine protection functions (e.g., turbine overspeed protection). The turbine-generator controller also provides trip and load rejection signals to the ATS and similar signals to the RPC to quickly open the bypass valves. The controller receives both high reactor level and vacuum trip signals from the DL4a DPS (to ensure these trips occur in the right sequence with other control systems). The turbine-generator controller is involved in plant automation to allow the turbine to be rolled, synchronized, and initially loaded. The operator always deliberately puts the system into automated control and several adverse conditions and diagnostics takes the system out of automation.
8. Rod Control and Information System - Provides the control logic and positions control rods using their fine motion drive motors.

RC&IS stores at least two predefined control rod sequences from all control rods full in (shutdown) to all control rods full out (i.e., end of cycle power operation). The sequences are carefully designed to avoid exceeding thermal limits and to meet fuel operating guidelines concerns under expected conditions of load following and transient xenon. Once a sequence is selected, RC&IS is constrained to operate in that sequence, but as described for the blocking systems, RC&IS is supervised by blocking functions to ensure that the sequences

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

are followed, and no thermal limits or fuel operating guidelines are exceeded. The rod block functions are listed in Table 7.3-4.

In the automatic mode, RC&IS accepts insert and withdraw commands from Automatic Power Regulator, these commands are issued as required to achieve the desired reactor power. Rod motion is constrained to the RC&IS selected sequence and Automatic Thermal Limit Monitor (ATLM) and MRBM are always functioning to supervise and block control rod motion. Only the operator can put the RC&IS into the automatic mode, but certain plant events or diagnostics can take the system out of automation mode.

In the semi-automatic mode, the operator depresses the insert or withdraw pushbuttons in the MCR. Once momentarily depressed the RC&IS inserts or withdraws the rods until the next predefined rod position breakpoint and then stops. Rod motion is constrained to the selected sequence and the ATLM and MRBM are actively enforcing rod motion blocks if unacceptable conditions are detected.

In the manual mode, the operator can move rods out of sequence (until blocked by the ATLM or MRBM) but unlike the other modes, only single (as opposed to ganged) rod motion is allowed. At lower thermal powers, manual movement commands deviating from the selected sequence are blocked. At higher thermal power levels, manual movement commands deviating from the sequence are alarmed.

RC&IS moves the control rods to a predefined rod pattern (by insertion only) when a SCRRI demand is issued. The SCRRI rod pattern is chosen to achieve a specific reactor power.

The RC&IS provides position demands to the control rod motor controllers and drives which are in DL4a. The DL4a design features allow the FMCRDs to be commanded to full in or to be blocked by disabling rod withdrawal regardless of RC&IS status or RC&IS positioning demand signals. The design allows a DL2 system to give commands to a DL4a system since the DL2 system is supervised by the blocking systems and the RC&IS inputs to the FMCRD motor controllers can be bypassed if the blocking systems so dictate.

9. Nuclear Boiler/Containment Inerting System – The nuclear boiler function provides most of the reactor flux, level, pressure, temperature, and flow information. The Nuclear Boiler system only transfers the signals to those systems that acquire and use the monitored signals. The instruments providing the data are typically mounted on instrument racks outside containment or routed from the vessel to outside containment signal conditioning and data acquisition equipment. It also has control requirements for valves repositioned as the plant is brought from cold to full power or respond to unplanned transients. SC1 equipment cannot be controlled by the Safety Category 3 automation function and therefore provision is made to operate the valves manually by the operator (as prompted by the automation system). Non-Safety Class valves and SC3 valves can be operated by the automation system. The containment Inerting function provides monitoring and control for the generation and storage of nitrogen gas to do the initial containment fill and thereafter to make up for the small long-term losses. It also supplies nitrogen to any inside containment pneumatic valves.
10. Control Rod Drive Hydraulic Control – CRD Hydraulic Control normally provides for flow control of the FMCRD purging functions and charging of the HCU accumulators to provide for the reactor hydraulic scram. There are two controllers that control the redundant CRD pumps and valves. The failure of the CRD controller to maintain the charge of the accumulators is alarmed to alert the operator if any of the HCU pressures and levels are too low.
11. Shutdown Cooling System - Provides separate controllers for monitoring and control for the SDC system to remove decay heat from the reactor. There are two separate SDC controllers

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

that respectively operate the redundant SDC. The SDC controllers can receive commands from the Plant Automatic System or can be manually operated.

12. Containment Cooling System - Provides redundant monitoring and control of the redundant containment air handling units with chilled water cooling during normal operation or during an outage.
13. Isolation Condenser System – Provides monitoring and control for the ICS pool redundant heat exchangers and pumps and their associated valves.
14. Fuel Pool Cooling and Cleanup System - Provides monitoring and control for the fuel pool cooling system. The system includes redundant pumps and heat exchangers and filter demineralizers. The controller also includes logic to start a standby pump if an operating pump trips.
15. Plant Service, Closed Cooling Water and Chillers - Provides redundant monitoring and control for service water pumps and valves. It also provides monitoring and control to mixing valves in series and parallel with the closed cooling water heat exchangers to provide a constant cooling water temperature regardless of cooling load. The closed cooling water pumps, the heat exchangers, and their valves are all controlled and monitored. It also provides control of the chillers and associated air-cooling fans and water distribution system valves requiring control and temperature monitoring.
16. R20 Electric Systems - The R20 electric system is divided into two medium voltage busses and their derivative low voltage busses. Also included are the two standby diesel generators. The load breakers on the various medium voltage and low voltage busses are controlled by the system controllers for those loads but the feeder breakers, load centre breakers, motor control centre breakers and diesel generator breakers are controlled by the two R20 controllers. The electric system control is usually by operator manual action except for the diesel generator breakers which close automatically after a loss of associated buss power after the diesel has been started and reached rated speed. The R20 controllers also control the sequencing of the standby diesel generator loads to minimize operator involvement and prevent diesel overload and under or over voltage and frequency. The controller provides the ability to synchronize the standby diesels to an energized bus to allow periodic load testing of the diesel generator. The controllers also provide extensive monitoring of the electric system parameters and for the R20 UPS, battery chargers and batteries.
17. R20 Protective Relaying - The R20 protective relaying function uses digital protective relays that appropriately protect the R20 medium voltage and low voltage loads, busses, feeder circuits and the standby diesel generators. The relays are redundantly powered and trip to protect the associated circuits or equipment but the data source function for I&C is not critical to their operation. Typically, the relays have enough redundancy that plant operation is not adversely affected by single relay failures. The relays are also in the control system architecture because they are networked to the C20 BOP Segment and the source of most of the electrical system parameter measurements. The required I&C function of the R20 Protective Relaying system is to interface the relays through gateways as necessary so the various data are available to I&C to support operator displays and alarms of electrical system status and for advanced condition monitoring.
18. Reactor Building and Control Building HVAC - Provides redundant monitoring and control for automatic operation of the normal Reactor and Control Building HVAC requirements by controlling the redundant cooling of the various Reactor and Control Building air handling units, building intake and exhaust fans, and, if required, recirculating fans. The room and building temperatures are controlled by damper and chilled water control valve operation.



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

19. Process Radiation Monitoring - Provides radiation data to other systems for trips but also provides data for operator monitoring, alarming and recording. The system controller provides for the acquisition of the various process radiation measurements, alarms and diagnostics and transmission of these data to the C20 Nuclear Segment networks with appropriate isolation or using appropriate safety class equipment to generate and transmit the data.
20. Defense Line 3 Gateways - The gateways are used to bring data linked signals to the C20 Nuclear Segment networks. The DL3 signals are isolated with SC1 isolators at the source. The two gateways are DL2 / SC3 components and have three major functions. The first function provides another layer of isolation. This isolation functions also prevents communication back to the C10 sources, which would not accept it in any case. The second function is to provide protocol conversion of the C10 signals to be compatible with the plant networks and package the data into messages for specific C20 controllers. The third function is to accept data links from non-SC1 sources that are input to the C20 Nuclear Segment, for example the data link from the DL4a DPS isolated diagnostics.
21. Automatic Thermal Limit Monitor - Provides redundant protection against critical power ratio (a fuel heat flux to flow consideration) and linear heat generation rate (maximum heat flux through the fuel clad) design limits. This online monitoring of thermal limits and fuel operating guidelines uses LPRM and APRM data sent via qualified isolation devices. The fuel operating guidelines keep the fuel duty per node below previously conditioned power levels to prevent fuel failures. Operation above the preconditioned envelope per node is constrained to slow power changes. The ATLM rod blocks are automatically operative at higher reactor powers. The ATLM receives online LPRM data and uses it with previously stored and periodically updated 3D power distribution and RC&IS data for current rod positions and the next selected rods to be moved to predict the resulting thermal limits and nodal powers for the fuel operating guidelines. The ATLMS are connected to the plant networks.
22. Multi-Channel Rod Block Monitor - Provides redundant protection against critical power ratio and linear heat generation rate design limits. The MRBM rod blocks are automatically operative at higher reactor powers. The MRBM uses LPRM and APRM data redundant pair of processing units with a real time operating system that each monitor LPRMs and APRMs sent via qualified isolation devices and control rod status from RC&IS. The MRBMs operate after receiving a signal from RC&IS that rods have been selected for motion. The MRBMs, using the real time LPRM information received from the gateways the MRBMs identify the LPRMs around the selected rods and normalize their average to a nominal value. As the selected rods are withdrawing, the LPRM average around the selected rods increases in value. If the ratio of the withdrawing average to the initial average increases by too great an amount, it indicates that the local powers have increased by an amount that would violate fuel thermal limits. Under those circumstances, the MRBMs issue a rod block. The MRBMs are connected to the plant networks.
23. Wide-Range Neutron Monitoring - The WRNM function provides redundant monitoring of the fixed neutron monitoring detectors in BWRX-300 core. The detectors are distributed radially in the core at fixed heights with at least three detectors in each quadrant. The detectors can be bypassed but the bypass switches are arranged such that only one detector can be bypassed at a time in any quadrant. Each detector is sensitive to neutrons from below criticality to greater than 100% thermal power. Their absolute value is indicative of reactor power. The WRNM detectors reactor power indication is not accurate given how few detectors are in the core, but they can indicate that the core has been successfully shutdown post rod insertion and for controlling reactor flux rate of change during a reactor startup. The average of the detectors can be related to thermal power by the comparison to APRM data. This relationship is continuously adjusted to ensure that there is always an overlap in the 1 – 10%

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

power region. Although the very localized WRNM absolute values are not required to be accurate, the main use of the detector data is to determine reactor period per detector. These period data are representative of the whole core and are used for monitoring and by Automatic Power Regulator. Both the operator and Automatic Power Regulator use the data when starting the reactor. The WRNMs are connected to the plant networks.

24. Stability Monitoring - Provides redundant operator alarms and takes the plant out of automation mode to ensure potential oscillations are not made worse by the reactor power control system. The design of the BWRX-300 and its core design make it less susceptible to core power oscillations, which are addressed as a fuel and core design criteria. The redundant Stability Monitoring functions check the LPRM signals sent via qualified isolation devices and oscillations and, when warranted, initiate an alarm and automation trip.
25. Safety Parameter Display System (SPDS) – The SPDS provides big picture overview display of plant parameters during transients and accidents. The SPDS terminology is used for consistency with regulatory documents; however, the BWRX-300 SPDS is actually functionality that is included with the C20 SC3 equipment. The BWRX-300 SPDS presents sufficient information on safety-critical parameters and includes display and trending of the parameters associated with the emergency procedure guidelines for the plant. The specific displays are developed in accordance with the Human Factors Engineering process described in Chapter 18. The displays cover the full range expected for the safety-critical parameters, track data trends, indicate when process or safety limits are being approached or exceeded, and display the status of safety systems. The SPDS is designed to use the same accident monitoring information provided to the C10 and C20 operator displays and is available in the Emergency Response Facilities described in Chapter 19
26. Operational Limits and Conditions Monitor – Provides surveillance test support and monitors and alarm deviations from plant OLC requirements for those requirements where appropriate plant signals are available. The redundant OLC Monitors have no signals of their own but instead have access to the signals on the plant networks including three divisions of C10 signals available to the DL2 Nuclear Segment through isolated, unidirectional boundary devices and DL2 gateways. The OLC Monitors are designed to support online and automatic surveillance testing to meet OLC requirements. Redundant input signals from the C10 SC1 I&C divisions used for safety actuation are continuously compared by the OLC Monitors during operation and alarmed for deviations outside of established tolerance. Similarly, the correct functioning of the C10 SC1 digital I&C platforms are continuously monitored by self-testing features rather than by manual surveillance testing. Critical faults detected by the self-testing features trip the outputs of the platform and other faults are alarmed. The OLC Monitors also monitor and alarm for other faults detected by self-testing and loss of communication indicating health and operability from any of the C10 SC1 I&C divisional components and most TMR controllers in the other DLs. The OLC Monitors are a redundant pair of processing units with a real time operating system that each monitor available OLC data. The OLC Monitors communicate among themselves to indicate they are working and agree on their determinations. The OLC Monitors have their own self-diagnostics, and the loss of a OLC Monitor is alarmed.
27. Core Thermal Power/Flow Monitor – Provides reactor thermal power and core flow information using a redundant pair of processing units with a real time operating system that calculates reactor thermal power and core flow. The thermal power is calculated once per second and is made available plant wide over the plant networks. The lowest power levels are estimated from the WRNM system and the intermediate power levels from the low flow feedwater system and bypass valve position. The program seamlessly transitions between the calculation methodologies. In the normal power range, approximately 10 – 100% thermal power, the

calculation is done using a heat balance. There are no assumed inputs or constants, heat balance parameters are measured and, in most cases, redundant. The most important measurements, feedwater flow, feedwater temperature and reactor pressure (used to determine the feedwater and steam enthalpies that account for greater than 99% of the heat balance accuracy) are triply redundant. Alarms are provided if sensed parameters are missing or inconsistent.

Natural circulation core flow is calculated by a heat balance around the core inlet. The important inputs are feedwater flow and enthalpy, steam flow and enthalpy and core inlet enthalpy, the latter using the core inlet temperature measurements in the LPRM and GT strings. A signal representative of core differential pressure (whose square root is proportional to core flow) is continuously calibrated against the heat balance calculation done in steady state. The calibrated differential pressure measurement is used during reactor transients and additionally to determine if the calibration constant changes significantly which may signal a restriction.

The core thermal power/flow monitors have their own self-diagnostics, and the loss of a monitor results in an alarm.

28. Three-Dimensional Core Thermal Power Distribution Monitor - These redundant computers use the thermal power calculation, core flow calculation, isolated (from DL3 and DL4a) GTs and LPRM data and control rod position data to determine the core 3D power distribution. The program runs both periodically (typically every hour) and on demand from the operator or ATLM (discussed above). Thermal limits and other diagnostics are alarmed, and data are recorded in the plant historians. The 3-D core monitors are a redundant pair of processing units with a real time operating system that each provide core monitoring, fuel operating guidelines, and thermal limit information. The core monitors have their own self-diagnostics, and the loss of a monitor results in an alarm.
29. Plant Alarm System - Provides aggregated process alarms and self-diagnostic data from the plant control and monitoring systems over the plant networks. The alarms and data are prioritized and filtered by plant mode and events. The alarm system also provides annunciator displays integrated into the MCR video displays. The purpose of the annunciator displays is to attract operator attention to specific display formats associated with the alarm. The operator can display currently active alarms, acknowledged alarms and "first in" alarms. The alarms are available to the operator and recorded in the plant historians. Alarm information is available in both control rooms and in the Site Management Centre. The alarm systems are redundant pairs of processing units that each monitor plant data. The alarms are available even with a single alarm system failure.
30. Plant Historians and Monitoring – Provides archiving of the plant control, alarm, and monitoring system information. The data is recorded at least once a second, depending on the parameter, (e.g., flux and pressure) some are recorded at higher rates and (e.g., temperature) some are recorded at lower rates. The recording capacity is for longer than a fuel cycle. The historians allow archival storage and playback while the plant is online and support operator-initiated data trending on control room displays or report printouts. The historians are redundant pairs of processing units that each monitor plant data. The stored data is still available even with a single historian failure.
31. DPS Diagnostics – Provides capability to monitor the DPS status and diagnostic information. The digital diagnostic system is data linked through the gateways such that the DPS signals can be monitored, alarmed, and recorded. For example, the OLC Monitors can perform continuous channel deviation checks on the DPS input signals, compare the DL4a and DL2 splitter outputs and monitor DPS power supplies.

32. Fiber Optic Relay Panels - Provides capability to send data across DL boundaries when the use of data links and networks is not indicated. These non-software panels containing optical fiber transmitters and receivers can transmit either analog or discrete contact signals.

The setpoints for C20 ATS actuation logic are determined with the setpoint methodology defined in IEC 61888 using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the ATS equipment (e.g., sensors and processing units). DL2 is designed to prevent or mitigate anticipated operational transients before either DL3 or DL4a action is required. As such, for common physical parameters measured by the DLs, the DL2 setpoint would be designed to act first, since its design basis is to mitigate/prevent DL3 from needing to respond.

### **7.3.3.3 Fundamental Design Properties in the System Design**

The C20 Defense Line 2 SC3 system includes DL1 properties that represent the quality measures implemented to minimize potential for failures and initiating events to occur in the first place, by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and D-in-D features are discussed in the following five subsections.

#### **7.3.3.3.1 *Qualified***

The C20 SC3 equipment is designed to operate in the environment that would be expected during both normal operations and anticipated off-normal conditions. The C20 Defense Line 2 Nuclear Segment is qualified to perform its intended functions and meets IEC 61513 requirements and the equipment standards defined in Section 7 in IEC 61226:2020 requirements for Class 3 systems. The C20 Defense Line 2 Nuclear Segment software is developed to IEC 62138 Category C requirements. Qualification addresses both hardware and software aspects of the C20 Defense Line 2 Nuclear Segment. The C20 Defense Line 2 Nuclear Segment design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The C20 Defense Line 2 Nuclear Segment equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The C20 Defense Line 2 Nuclear Segment qualification measures confirm the I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The C20 equipment is qualified for seismic interaction in accordance with IEC 60980. The C20 equipment is qualified for electromagnetic compatibility in accordance with IEC 61000-4 and IEC 61000-6-2.

#### **7.3.3.3.2 *Reliable***

The C20 Defense Line 2 Nuclear Segment has the required reliability to perform its intended functions. The main C20 hardware platform uses TMR controllers for the major plant control systems. It has an initial quantitative reliability target of 1E-6 to 1E-7 probability of failure per hour for systems in continuous operation and 1E-2 to 1E-3 probability of failure on demand for standby functions. The reliability analysis of the C20 Defense Line 2 Nuclear Segment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate.

The C20 Defense Line 2 Nuclear Segment process controllers are designed with TMR controllers to prevent random I&C component failures from causing plant transients. The TMR controller outputs are dual ported to the plant C20 Nuclear Segment network. Where a mechanical system has safety functions and is redundant (i.e., "A" and "B"), two controllers are furnished so the

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

redundant components may be operated separately, and the system is not affected by a single failure.

For the functions with TMR controllers, single controller failures (and some double failures) have no adverse effect on system or plant operation and failed controller parts may be replaced online. The controllers are a DL2 / SC3 design with a specific requirement that there be no failures more often than once per 100 years. The intent is that no random controller failure initiates an AOO. The reliability of the controller depends on final design and the formal analyses to determine failure rates are performed as part of the detailed plant design.

C20 Defense Line 2 Nuclear Segment monitoring functions are implemented on redundant processors for reliability. Normally both redundancies are online, and their data continuously compared. Any one redundancy can be bypassed at a time but, if bypassed, it takes the plant out of automation mode.

Design reliability assurance program and reliability, availability, and maintainability plan documents are used to quantify the required failure rates of the DCIS to assure plant safety and plant availability goals. Testing and performance assessment requirements for ATS are developed using the guidance of IEC 60671 and IEC 62385.

Extensive hardware and software diagnostics are provided for the TMR controllers to provide for operator monitoring and alarms.

The feedwater temperature and level controller input signals are triplicated. No signal is derived from any other DL.

The C20 DL2 functions for ATLM, MRBM, WRNM, Stability Monitoring, SPDS, TSM, Core Thermal Power/Flow Monitor, 3-D Core Monitor, Plant Alarm System, and Plant Historians and Monitoring are provided with redundant UPS and power feeds.

#### **7.3.3.3.3 Robust**

Robustness of the C20 Defense Line 2 Nuclear Segment is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust C20 Defense Line 2 Nuclear Segment design reflects the use of design methods and adherence to engineering best practices to ensure that its functions are achieved for the specified conditions.

The C20 Defense Line 2 Nuclear Segment has the required separation and independence to perform its intended functions. DL2 is independent from DL3. DL2 is also independent from DL4a to an extent that is practicable in that shared sensors are not credited with mitigation of the same PIE. DL2 and DL4a share support equipment. DL2 is designed to ensure that it cannot adversely affect DL3 or DL4a from fulfilling their protection functions. DL2 shares some signals with DL4a sent via qualified isolation devices (as described in Subsection 7.3.2.3.3). DL2 may use signals from DL3 sent via qualified isolation devices (as described in Subsection 7.3.1.3.3) for some control purposes. Plant operation is not adversely affected by the loss or inaccuracy of these individual signals. The C20 DL2 ATS functions do not use any sensors or actuators in common with DL3.

The DL2 equipment is located in two separate fire barrier rooms in the Control Building. The DL3 equipment is located in three separate divisional fire barrier rooms in the Reactor Building. The DL4a equipment is located in a separate fire barrier room in the Control Building.

There is no communication from ATS to DL3 and only rod block or motor run-in initiation (dry contact) communication to the DL4a ERIPs. There is no direct connection between DL3 and the Automatic Power Regulator. The Automatic Power Regulator does use APRM signals from DL3 for power feedback, but it also uses GTs signals and the core thermal power computers for similar

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

feedback. These signals are continuously compared, and any discrepancy takes the plant out of automation mode to prevent any power signal from misleading the control system. Other than the APRM signals, DL3 / SC1 and the Automatic Power Regulator controller have and no sensors or actuators in common. There are no DL2 commands or communication to DL3. The C20 physical separation and signal isolation devices meet the requirements of IEC 60709.

The C20 Defense Line 2 Nuclear Segment has the required fail-as-is design features to perform its intended functions and avoid spurious actuation. The fail-as-is energize to actuate design is used to prevent lesser classified systems (e.g., ATS) from creating unnecessary challenges to plant safety (e.g., spurious actuations for expected failures) requiring action by DL3. Where software is used for diagnostics, DL2 diagnostics always include watchdog timers, sensor range checks and power supply, communications, and actuator monitoring. The ATS is designed to energize its outputs to cause a trip or initiation. Each output is produced separately by a 2-out-of-3 discrete, or median select analog vote from each of the three controllers. The scheme is used to provide reliable trips and avoid inadvertent trips.

The blocking system functions of DL2 (i.e., ATLM, MRBM, and WRNM) are redundant and are used to supervise the control systems and plant conditions. Either of the redundant blocking pair can block rod motion. One of the redundant pair can be bypassed for maintenance activities. The bypass condition automatically takes the plant out of automation mode, which halts automatic control rod motion under command of the Plant Automation System. The remaining component of the blocking pair can still block rod motion in any plant mode. If both blocking pairs are not available semi-automatic and manual control rod motion is halted until at least one is available. The blocking pairs alarm and block if they lose communication with the gateways or RC&IS, cannot receive an update from the core 3D core thermal power distribution monitoring program, or if their internal data do not agree.

Either WRNM can cause a rod withdrawal block (unless bypassed) for a short period and either can cause a hydraulic scram via the ATS for a shorter period. The scram is only available when the Reactor Mode Switch is not in RUN (typically below 15% thermal power as measured by the APRMs). If no WRNM detector is available in any core quadrant, rod motion is prohibited except in manual mode. If no WRNM detector is available in any core quadrant during refueling activities, then fuel movement is halted.

Any one Stability Monitoring function can be bypassed at a time, but if bypassed, automatic control rod motion under command of the Plant Automation System is halted. If both Stability Monitoring functions are not available semi-automatic and manual rod control is halted until at least one Stability Monitoring function is available (later plant design and analysis may conclude that no power restrictions are required). Either Stability Monitoring function (unless bypassed) can cause an alarm and automation trip.

The C20 Defense Line 2 Nuclear Segment is predictable when performing its intended functions. Where ATS or a reactor control system TMR controller uses three signals from DL4a splitters, a fourth DL2 signal is used for the ATS. Typically, these parameters include containment pressure, reactor level, reactor pressure, and condenser vacuum scrams. Each ATS controller and reactor control system validates the analog signal with a range and consistency check, compares each of the four signals to a common setpoint, and then indicates the analog value and trip status. For example, the reactor level and pressure controller analog validation algorithms use the three splitter signals from DL4a, and the fourth signal dedicated to DL2 functions to make a single failure proof control signal that functions even if the splitter signals are lost:

- If four signals are valid, use average of two median signals
- If three signals are valid, use median signals

- If two signals are valid, use average of two signals
- If only one signal is valid, use signal

For trip signals derived from analog inputs the DL2 validation algorithms functions:

- If four signals are valid, use 2-out-of-4 logic
- If three signals are valid, use 2-out-of-3 logic
- If two signals are valid, use 2-out-of-2 logic
- If only one signal is valid, do not trip

The scheme is used to provide reliable trips and avoid inadvertent trips. The simple logic for the ATS TMR controller is faster and acts before the three asynchronous DL3 divisional controllers voting communications and faster than the analog DPS.

The C20 Defense Line 2 Nuclear Segment provides diversity to support the D-in-D concepts. The DL2 C20 hardware and software platforms are diverse from DL3. Aside from common but isolated power supplies and common support equipment like HVAC, DL4a protection functions are mostly analog and any DL4a supporting software is also diverse from DL2 (and DL3). It can be stated that DL2 software and logic is independent of DL4a. The processing units used for the control rod blocking functions are implemented in equipment diverse from the DL3 equipment if it is credited as a backup in the safety analyses. The turbine overspeed protection control logic uses two diverse and independent systems to provide sufficient reliability to eliminate the risk of turbine missile generation and to avoid the need for a mechanical overspeed trip. These turbine overspeed protection systems use TMR control architecture to eliminate random hardware vulnerabilities that could result in spurious turbine trips.

#### **7.3.3.3.4    *Secure***

The C20 Defense Line 2 Nuclear Segment adequately addresses security to limit cyber security vulnerabilities and ensure the system can perform its intended functions.

The C20 Defense Line 2 Nuclear Segment design incorporates features to support establishment of a secure operational environment. It is installed and maintained in accordance with the station administrative procedures and control of access programs.

The BWRX-300 network switches interconnect the C20 segments have security features, as described in Subsection 7.2.3. The communications on the nuclear Unit Data Highways are rigorously controlled and monitored and information sent off-site is through unidirectional boundary devices, as described in Subsection 7.2.4.

#### **7.3.3.3.5    *Defence-In-Depth***

The C20 Defense Line 2 Nuclear Segment provides anticipatory trip functions to support the D-in-D concepts. Examples of anticipatory trips include an ATS scram on loss of off-site power and is initiated before the DL3 and DL4a systems scram the plant on reactor low level on loss of the feedpumps or a high flux resulting from a reactor isolation on loss of vacuum. Another example is the ATS scram on a turbine trip or load rejection before the DL3 and DL4a systems scram the plant on reactor high flux or pressure; this also mitigates the adverse thermal limit response because the reactor is scrammed before the pressurization wave impacts the reactor. The DL2 C20 hardware and software platforms are diverse from DL3. The C20 Defense Line 2 ATS actuation relays shown in Figure 7.3-4 are configured as energize to actuate whereas C10 SC1 I&C load drivers are configured as deenergize to actuate, which provides functional diversity in these actuation circuits.

Diversity measures are incorporated within the C20 Defense Line 2 Nuclear Segment design as an additional level of protection for potential systematic faults caused by design and implementation defects when equipment it is credited to backup' in the safety analyses. Rod blocks are hard wired to RC&IS. The Stability Monitoring function is provided for indication only as additional D-in-D.

The use of a fourth DL2 signal provides immunity a loss of control signal due to the postulated CCF of the DL4a signal splitters.

DL2 can initiate hydraulic and motor run-ins, initiate isolations, and operate the ICS, no DL2 equipment can prevent a scram, isolation, or ICS initiation actuated from DL4a or DL3. Similarly, DL3 or DL4a cannot adversely affect DL2 operation. If DL2 does not send out its anticipatory trip commands, plant safety is ensured by DL3 or DL4a.

Although DL4a, the DL2 Nuclear Segment, and the C30 BOP Segment are powered by the same pair of UPS with 72-hour battery backup, the power supply is redundant, and a single failure does not affect either DL or segment. The UPS and the battery chargers can be powered by off-site power or either of the standby diesel generators. The DLs use double breakers or fuses to separate the DLs and to separate the C20 Nuclear and C20 BOP Segments such that a fault in a DL or segment does not adversely affect the other DLs. No DL4a or DL2 power is shared with DL3.

#### **7.3.3.4 Operator Interface**

The C20 data are available on their appropriate displays. The layout and content of the displays are designed as part of the HFE program of activities integrated into the BWRX-300 design, as described in Chapter 18, Section 18.3. C20 has its own displays in both the MCR and SCR and locally in the C20 equipment rooms. Each divisional display is dedicated to monitoring or controlling its associated division. The C20 displays also include an alarm system to provide for operator awareness and to prompt to the displays containing further information relating to the alarm. C20 system bypass conditions are alarmed in the MCR. The displays are menu driven and it is possible to switch between related displays without going through the main menu.

Some C20 monitoring signals provide necessary information to achieve or verify correct plant response to transients and accidents. The necessary signals are determined through HFE analysis and applicable regulatory requirements and are referred to as accident monitoring signals.

The C20 signals are alarmed to prompt the operator to an appropriate display (e.g., range check problems or alarm setpoints exceeded). The C20 signals and the various C20 components are alarmed for self-diagnostics.

The C20 SC3 system acquires accident monitoring information associated with Type B and C accident monitoring variables from C10 SC1 I&C through isolated interfaces. It also acquires accident monitoring information associated with Type D, E, and F accident monitoring variables from C20 and C30. The aggregated C20 displays of accident monitoring information are provided to the MCR and Site Management Centre located in the Control Building and the SCR located in the Reactor Building in the current concept design. The Type D, E, and F are defined in IEC 63147. Type D variables provide information to accident management personnel to verify safety system status and that indicate the performance of safety systems, required auxiliary support features, and other systems necessary to achieve and maintain a safe shutdown condition. Type E variables provide information to accident management personnel to monitor radioactive releases identified pathways and environmental conditions used to determine the impact of releases in the plant environs and radioactivity levels in the control rooms and selected plant areas where access may be required for plant recovery. Type F variables provide information



that indicates fuel damage and the effects of fuel damage to support accident management response to severe accidents. A preliminary list of the Type D, E, and F accident monitoring variable is provided in Table 7.3-2.

The C20 and C30 DCIS rooms, which include I&C equipment for DL2, some Non-Safety Class equipment, and associated electrical support equipment are SC3 and located in the Control Building. The MCR is used by plant operators unless conditions require relocation to the SCR. The Site Management Centre is used by accident management personnel. The Control Building provides the necessary seismic ruggedness and habitability capabilities to support the expected post-accident conditions with a SC3 level of quality and reliability for required protection functions.

The operator interface HSI for the C20 Nuclear Segment of DL2 is described in Sections 7.5 and 7.6 for the MCR and SCR, respectively.

#### **7.3.3.5 Compliance Alignment**

Table 7.3-7 shows how the C20 Defense Line 2 Nuclear Segment aligns to meets the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603.

#### **7.3.4 System C30 – Non-Safety Class / Balance of Plant Segment**

Section 7.2.1 describes how the C20 network is split into two segments: Nuclear and BOP. This section addresses the C30 BOP controllers that feed the C20 BOP Segment, which contains the control and monitoring systems associated with power generation and support systems

##### **7.3.4.1 Systems Architecture**

The functional architecture for the C30 BOP controllers is shown in Figure 7.3-13. Depending on final DCIS design and processor loading, functions may be combined onto common or shared controllers.

The C30 controllers in the BOP Segment are either TMR controllers or “packaged systems” (typically embedded controllers) provided by an equipment vendor.

Control signals needed between controllers for their functions are point-to-point wire or optical fiber. The signals needed by a controller for its functions are radially connected to the controller and do not use plant networks.

##### **7.3.4.2 System Design Bases and Associated Safety Functions**

The design basis of the C30 controllers is to provide for the control and monitoring of the Non-Safety Class power generation and support equipment. The C30 I&C system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states in order to ensure that adequate information can be obtained on plant status. This equipment is not credited for protection functions although they may be available for beyond design basis and severe accident scenarios.

The main C30 functions are:

1. Condensate and Feedwater – Provides monitoring and control for the condensate and feedwater systems as the plant is brought from cold to rated power conditions. The condensate and feedwater functions use TMR controllers that are dual ported to the plant C20 BOP Segment network.
2. Feedwater Heater Drains and Extraction Steam – Provides monitoring and active heater level control functions to prevent heater isolation, turbine trip and to provide optimum heat transfer.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

The feedwater heater drains, and extraction steam function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.

3. Condenser – Provides monitoring and “batch” process functions as the plant is brought from cold to rated power. The condenser also requires extensive monitoring for both condition monitoring and investment protection purposes. The condenser function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
4. Turbine Auxiliaries – Provides control of several auxiliary systems that are controlled for normal plant operation. The turbine auxiliaries also require extensive monitoring for both condition monitoring and investment protection purposes. The turbine auxiliary function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
5. Generator Auxiliaries – Provides control for the stator cooling system and may also provide the interface to the generator exciter to provide generator voltage regulation and reactive power (dispatch as requested by the load dispatcher). The generator auxiliaries also require extensive monitoring for both condition monitoring and investment protection purposes. The generator auxiliary function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
6. Moisture Separator Reheater – Provides reheater stage and the moisture separator level control reheater exit steam temperature control. The Moisture Separator Reheater System (MSR) also requires extensive monitoring for both condition monitoring and investment protection purposes. The MSR function uses TMR controllers that are TMR and dual ported to the plant C20 BOP Segment network.
7. Turbine Bypass Valves – Provides position feedback linear variable differential transformers and servo valves are part of the mechanical turbine bypass valve system. The bypass valve controller is the TMR RPC, but the monitoring is part of the BOP bypass valve control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
8. Reactor Water Cleanup System – Provides monitoring and control for the Reactor Water Cleanup System (CUW) flow in normal operation to maintain required system flow rates and protect the cooling water temperatures of the regenerative heat exchanger. The CUW function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
9. Circulating Water – Provides monitoring and control of the cooling water flow to the main condenser. The CW function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
10. Intake Structure – Provides monitoring and control for racks and travelling screens to prevent debris from entering the water systems and associated HVAC to maintain the intake building environment. The intake structure function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
11. Turbine and Radwaste Building HVAC – Provides monitoring and control for automatic operation of the plant Non-Safety Class chillers. The Non-Safety Class HVAC control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
12. Service Air and Instrument Air and Plant Gas Systems – Provides monitoring and control for the Service Air and Instrument Air systems. The Service Air, and Instrument Air control function, uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
13. Condensate Polishing System – Provides control and monitoring of the Condensate Polishing System, which consists of multiple vessels for filtration and demineralization of the condensate water before injection into the reactor. The system is designed to be highly reliable for

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

maintaining reactor chemistry control and to prevent inadvertent operation that could disrupt the condensate/feedwater flow to the reactor vessel. The polishing system vessels are automatically sequenced and regenerated/backflushed by the controller as chemistry dictates. Additionally, the controller provides vessel flow control to provide equal flow such that the resins and filters are not simultaneously depleted. The Condensate Polishing System control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.

14. R30 Electric System – Provides extensive monitoring of the electric system parameters and for the R30 UPS, battery chargers and batteries. The R30 control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
15. R30 Protective Relaying – The R30 protective relaying function uses digital protective relays that appropriately protect the R30 medium voltage and low voltage busses and loads, feeder circuits, main and unit auxiliary transformers and the main generator. Where a single failure can cause a plant scram, the main generator, main transformer, and unit transformer are protected with a 2-out-of-3 relaying scheme. The relays are also in the control system architecture because they are networked to the C20 BOP Segment and the source of most of the electrical system parameter measurements. The required I&C function of these systems is to interface the relays through gateways as necessary so the various data are available to I&C to support operator displays and alarms of electrical system status and for advanced condition monitoring.
16. Condensate Storage and Transfer – Provides monitoring and control for the Condensate Storage and Transfer system to store condensate quality water for power generation and transfer the water to its various uses in the power plant. The Condensate Storage and Transfer system control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
17. Equipment and Floor Drains System – Provides monitoring and control for the plant sumps and drains to collect and release or process depending on the radioactivity and chemistry of the contained water. The equipment and floor drains system control function use TMR controllers that are dual ported to the plant C20 BOP Segment network.
18. Potable Water and Sewage – Provides monitoring and control for the capability to ensure that there is at least a one-week supply of stored water potable available in each control room. The potable water and sewage control function uses TMR controllers that are dual ported to the plant C20 BOP Segment network.
19. Hydrogen Water Chemistry and Noble Metal Injection - Provide a dedicated controller for skid mounted mechanical components that mechanically inject chemicals into the feedwater flow to the reactor. Hydrogen injection is continuous at power and proportional to feedwater flow. Noble metal is injected at commissioning and then periodically over the plant lifetime. The hydrogen water chemistry and noble metal control function uses TMR controllers that are dual ported to the C20 BOP Segment network.
20. Liquid and Solid Waste Management and Offgas – Provides monitoring and control for the offgas system and the liquid and solid waste systems. The offgas controller involves both motor-operated and modulating valves to operate the steam air ejectors and steam dilution systems. The controller provides the systems' monitoring and control capability, specifically included is radiation monitoring before radioactive material or gas is released. The liquid and solid waste systems provide batch controller functions. The liquid and solid waste management and offgas control function uses TMR controllers that are dual ported to the C20 BOP Segment network.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

21. BOP Gateways – The BWRX-300 may have many packaged systems bought from vendors instead of being designed by GEH. These vendor systems typically include embedded control systems. These vendor systems are integrated into the plant DCIS for both monitoring and control purposes. Each packaged system is connected to the DCIS networks through a gateway to minimize cyber security vulnerabilities. The gateways are designed to provide any needed protocol conversion and cyber security data flow features that limit what the embedded controller can do.
22. Fire Protection (Packaged) – Packaged system that meets local fire safety standards. The system is self-contained with its own operator interfaces (see Chapter 9A, Section 9A.6). A gateway is provided to interface to the plant DCIS so that the operator can monitor the system, obtain immediate fire alarm location and actuation information, and dispatch an appropriate response before the fire protection system console can be manned.
23. Water Treatment (Packaged) – Packaged plant that inputs raw water and makes demineralized water (see Chapter 9, Subsection 9A.9.5). The packaged plant embedded controller controls its pumps and demineralizers and backwashing/regeneration and operates in either batch or continuous mode.
24. Clock (Packaged) – The DCIS includes a set of redundant Global Positioning System satellite synchronized clocks. The main use of the clocks is to distribute network time through gateways to the plant control systems, historians, and electrical system digital protective relays so that signals from different sources are synchronized when recorded and for playback.
25. Equipment Condition Monitoring (Packaged, if provided) – The Equipment Condition Monitoring equipment consists of a central console that receives data from special remote multiplexing units located near power generation rotating machinery including the main turbine, motors, pumps, and fans.
26. Seismic Monitoring System (Packaged) – The Seismic Monitoring System is a packaged system (see Chapter 3, Subsection 3.3.1). It has its own internal UPS, an operator interface, and recording capability. There are two and three axis accelerometers located in various plant locations, including the yard. These are directly wired to the Seismic Monitoring System cabinet. Whenever an accelerometer signal exceeds a predefined value, the Seismic Monitoring System begins recording accelerometer data. The Seismic Monitoring System can also input a contact alarm into the plant DCIS in addition to a datalink through a gateway to the DCIS networks. The alarm alerts plant staff to go to the Seismic Monitoring System operator interface to determine if there is plant damage. The intent of the system is, post-earthquake, to allow the operator or plant engineering staff to determine if the earthquake was above or close to the design basis earthquake. This informs the decision of whether to shut down the plant or how much inspection of plant equipment is required. The Seismic Monitoring System is connected to the plant networks only for alarming purposes.
27. Area Radiation Monitoring System (Packaged) – The Area Radiation Monitoring System (ARMS) is a packaged system with detectors at many plant locations accessible to personnel and subject to potential high radiation. The ARMS cabinet has an operator interface and a datalink to the plant DCIS. There are local radiation detectors mounted in various plant locations with the ability to provide both radiation data and predefined high radiation alarms. (See Chapter 12, Subsection 13.3.14.4)
28. Meteorological Monitoring System (Packaged) – The Meteorological Monitoring System consists of a site tower containing wind speed, wind direction, temperature, and barometric pressure instruments. The data are used with environmental monitoring radiation release

data to predict radiation plume magnitude, direction, and dispersion. (See Chapter 2, Subsection 2.11.8)

29. Environmental Monitoring System (Packaged) – The Environment Monitoring System consists of sensors located at or near site boundaries to monitor plant releases. Radiation releases are typically measured but the system may be used to monitor other parameters required by environmental permits. (See Chapter 20, Subsection 20.11)

#### **7.3.4.3 Fundamental Design Properties in the System Design**

The C30 Non-Safety Class system properties that represent the quality measures implemented to minimize potential for failures and initiating events to occur in the first place by the use of conservatism in design and analyses. These properties of qualification, reliability, robustness, security, diversity, and D-in-D features are discussed in the following five subsections.

##### **7.3.4.3.1 *Qualified***

The C30 controllers are Non-Safety Class. The C30 equipment is designed for to operate in the environment that would be expected during normal operations. The C30 controller design and manufacturing processes are of sufficient quality to ensure that I&C systems can reliably perform their design basis functions. The C30 controller equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The C30 controller qualification measures confirm that the I&C systems and equipment are capable of reliably performing the design functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The C30 controller equipment is assessed as a potential hazard if located near safety classified equipment and, if necessary, qualified for electromagnetic compatibility in accordance with IEC 61000-4 and IEC 61000-6-2 for emissions only.

##### **7.3.4.3.2 *Reliable***

The C30 controllers have the required reliability to perform their intended functions. The reliability analysis of the C20 BOP Segment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate. Separately, the C30 controllers have the required reliability to perform their intended functions.

C30 BOP Segment functions are implemented with TMR controllers where appropriate to ensure that single controller failures have no adverse effect on system or plant operation. The controllers have a specific reliability requirement goal that there be no failures more often than once per 100 years. The intent is that no random controller failure initiates an AOO. The reliability of the controller depends on final design and the formal analyses to determine failure rates are performed as part of the detailed plant design.

Extensive hardware and software diagnostics are provided for the TMR controllers to provide for operator monitoring and alarms. Failed TMR controller parts may be replaced online.

The main C30 controller platforms control and monitoring design supports both local and remote data acquisition, the latter with optical fiber or wire. The intent is that the failure of a C30 controller performing Non-Safety Class control functions never initiate an AOO. The platforms in this category have quantitative reliability targets of 1E-6 to 1E-7 probability of failure per hour for systems in continuous operation and 1E-2 to 1E-3 probability of failure on demand for standby functions.

#### **7.3.4.3.3 Robust**

Robustness of the C30 controllers is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust C30 controller design reflects the use of design methods and adherence to engineering best practices to ensure that its functions are achieved for the specified conditions.

The C30 controllers have no direct connections (i.e., commands or communication) with the DL3 / SC1 systems (C10).

The C30 controller equipment is in different rooms than the C20 Nuclear Segment rooms and physically separated from DL3 and DL4a rooms. The C30 controller physical separation and signal isolation devices meet the requirements of IEC 60709. Vendor supplied Non-Safety Class functions are implemented on a C30 hardware and connected through a gateway.

#### **7.3.4.3.4 Secure**

The C30 controller design incorporates features to support establishment of a secure operational environment. This equipment is installed and maintained in accordance with the station administrative procedures and control of access programs.

The BWRX-300 network switches that interconnect the C20 BOP Segment have security features, as described in Subsection 7.2.3. The communications on the BOP Unit Data Highways are rigorously controlled and monitored and information sent off-site is through unidirectional boundary devices, as described in Subsection 7.2.4. The vendor packaged C30 controllers are interfaced to the C20 BOP Segment through gateways to provide communication protocol conversion and eliminate cyber security vulnerabilities.

#### **7.3.4.3.5 Defence-In-Depth**

The C30 BOP Segment I&C platform equipment no specific diversity requirements since it performs no credited protection functions.

#### **7.3.4.4 Operator Interface**

The operator interface HSI for the C30 BOP Segment is described in Section 7.5 for the MCR.

#### **7.3.4.5 Compliance Alignment**

Table 7.3-8 shows how the Non-Safety Class C30 BOP Segment aligns to meet the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603.

### 7.3.5 References

- 7.3-1 CNSC Regulatory Document REGDOC-3.6, "Glossary of CNSC Terminology."
- 7.3-2 IEC 63147, "Criteria for accident monitoring instrumentation for nuclear power generating stations," International Electrotechnical Commission.
- 7.3-3 IEC 61888, "Nuclear power plants – Instrumentation important to safety – Determination and maintenance of trip setpoints," International Electrotechnical Commission.
- 7.3-4 IEC 60987, "Nuclear power plants – Instrumentation and control important to safety – Hardware requirements," International Electrotechnical Commission.
- 7.3-5 IEC 61513, "Nuclear power plants – Instrumentation and control important to safety – General requirements for systems," International Electrotechnical Commission.
- 7.3-6 IEC 61226, "Nuclear power plants – Instrumentation, control and electrical power systems important to safety – Categorization of functions and classification of systems," International Electrotechnical Commission, April 2020.
- 7.3-7 IEC 60880, "Nuclear power plants - Instrumentation and control systems important to safety - Software aspects for computer-based systems performing category A functions," International Electrotechnical Commission.
- 7.3-8 IEC/IEEE 60780-323, "Nuclear facilities – Electrical equipment important to safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 7.3-9 IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 7.3-10 IEC 61000-4-2, "Electromagnetic compatibility (EMC) – Part 4-2: Testing and measurement techniques – Electrostatic discharge immunity test," International Electrotechnical Commission.
- 7.3-11 IEC 61000-4-3, "Electromagnetic compatibility (EMC) – Part 4-3: Testing and measurement techniques – Radiated, radio-frequency, electromagnetic field immunity test," International Electrotechnical Commission.
- 7.3-12 IEC 61000-4-4, "Electromagnetic compatibility (EMC) – Part 4-4: Testing and measurement techniques – Electrical fast transient/burst immunity test," International Electrotechnical Commission.
- 7.3-13 IEC 61000-4-5, "Electromagnetic compatibility (EMC) – Part 4-5: Testing and measurement techniques – Surge immunity test," International Electrotechnical Commission.
- 7.3-14 IEC 61000-4-6, "Electromagnetic compatibility (EMC) – Part 4-6: Testing and measurement techniques – Immunity to conducted disturbances, Induced by radio-frequency fields," International Electrotechnical Commission.
- 7.3-15 IEC 61000-4-8, "Electromagnetic compatibility (EMC) – Part 4-8: Testing and measurement techniques – Power frequency magnetic field immunity test," International Electrotechnical Commission.
- 7.3-16 IEC 61000-4-9, "Electromagnetic compatibility (EMC) – Part 4-9: Testing and measurement techniques – Impulse magnetic field immunity test," International Electrotechnical Commission.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

- 7.3-17 IEC 61000-4-11, "Electromagnetic compatibility (EMC)- Part 4-11: Testing and measurement techniques - Voltage dips, short interruptions and voltage variations immunity tests," International Electrotechnical Commission.
- 7.3-18 IEC 61000-4-12, "Electromagnetic Compatibility (EMC) - Part 4-12: Testing and measurement techniques - Ring wave immunity test," International Electrotechnical Commission.
- 7.3-19 IEC 61000-4-13, "Electromagnetic compatibility (EMC) - Part 4-13: Testing and measurement techniques - Harmonics and interharmonics including mains signalling at a.c. power port, low frequency immunity tests," International Electrotechnical Commission.
- 7.3-20 IEC 61000-4-14, "Electromagnetic compatibility (EMC) - Part 4-14: Testing and measurement techniques - Voltage fluctuation immunity test for equipment with input current not exceeding 16 A per phase," International Electrotechnical Commission.
- 7.3-21 IEC 61000-4-16, "Electromagnetic compatibility (EMC) - Part 4-16: Testing and measurement techniques - Test for immunity to conducted, common mode disturbances in the frequency range 0 Hz to 150 kHz," International Electrotechnical Commission.
- 7.3-22 IEC 61000-4-18, "Electromagnetic compatibility (EMC) – Part 4-18: Testing and measurement techniques – Damped oscillatory wave immunity test," International Electrotechnical Commission.
- 7.3-23 IEC 61000-4-19, "Electromagnetic compatibility (EMC) – Part 4-19: Testing and measurement techniques – Test for immunity to conducted, differential mode disturbances and signaling in the frequency range 2 kHz to 150 kHz at a.c. power ports," International Electrotechnical Commission.
- 7.3-24 IEC 61000-4-28, "Electromagnetic compatibility (EMC) – Part 4-28: Testing and measurement techniques - Variation of power frequency, immunity test for equipment with input current not exceeding 16 A per phase," International Electrotechnical Commission.
- 7.3-25 IEC 61000-4-31, "Electromagnetic compatibility (EMC) – Part 4-31: Testing and measurement techniques – AC mains ports broadband conducted disturbance immunity test," International Electrotechnical Commission.
- 7.3-26 IEC 61000-4-34, "Electromagnetic compatibility (EMC) – Part 4-34: Testing and measurement techniques – Voltage dips, short interruptions and voltage variations immunity tests for equipment with mains current more than 16 A per phase," International Electrotechnical Commission.
- 7.3-27 IEC 61000-6-2, "Electromagnetic compatibility (EMC) – Part 6-2: Generic standards – Immunity standard for industrial environments," International Electrotechnical Commission.
- 7.3-28 IEC 60671, "Nuclear power plants – Instrumentation and control systems important to safety – Surveillance testing," International Electrotechnical Commission.
- 7.3-29 IEC 62385, "Nuclear power plants – Instrumentation and control important to safety – Methods for assessing the performance of safety system instrument channels," International Electrotechnical Commission.



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

- 7.3-30 IEC 61500, "Nuclear power plants – Instrumentation and control systems important to safety – Data communication in systems performing category A functions," International Electrotechnical Commission.
- 7.3-31 IEC 60709, "Nuclear power plants – Instrumentation, control and electrical power systems important to safety – Separation," International Electrotechnical Commission.
- 7.3-32 IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 7.3-33 IEC 62138, "Nuclear power plants - Instrumentation and control systems important to safety - Software aspects for computer-based systems performing category B or C functions," International Electrotechnical Commission.

**Table 7.3-1: C10 SC1 I&C Functions and Initiation Signals**

Initiating Signal	Reactor Mode Switch Position
<b>DL3 Hydraulic Scram</b>	
HIGH RPV pressure [HP1]	RUN or STARTUP
LOW RPV pressure [LP1]	RUN
LOW RPV level [L3]	RUN or STARTUP
HIGH PRNM neutron flux [RUN setpoint]	RUN
HIGH simulated thermal power	RUN
HIGH containment pressure	RUN or STARTUP
HIGH PRNM neutron flux (STARTUP setpoint)	STARTUP
Line break indication (Main Steam Line (MSL), Feedwater (FW), or ICS)	RUN, STARTUP, or SHUTDOWN
<b>Isolation Condenser Actuation</b>	
HIGH RPV pressure [HP2] (ICS Train 1 actuation)	RUN, STARTUP, or SHUTDOWN
HIGH RPV pressure [HP3] (ICS Train 2 actuation)	RUN, STARTUP, or SHUTDOWN
HIGH RPV pressure [HP4] (ICS Train 3 actuation)	RUN, STARTUP, or SHUTDOWN
LOW RPV water level [L2]	RUN, STARTUP, or SHUTDOWN
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
Line break indication (MSL, FW, or ICS)	RUN, STARTUP, or SHUTDOWN
<b>RPV and Containment Isolation</b>	
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
<b>MSRIV and MSCIV Isolation</b>	
LOW RPV pressure [LP1]	RUN
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
Indication of a MSL break	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
<b>FWRIV and FWCIV Isolation</b>	
HIGH RPV Level [L9]	RUN, STARTUP, or SHUTDOWN
LOW RPV Level [L2]	RUN, STARTUP, or SHUTDOWN
Loss of Normal FW Flow	RUN, STARTUP, or SHUTDOWN

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Initiating Signal	Reactor Mode Switch Position
<b>Feedwater and SDC Isolation</b>	
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
<b>CUW Isolation</b>	
Indication of a CUW line break	RUN, STARTUP, or SHUTDOWN
<b>ICS Train Isolation (Train 1)</b>	
ICS train 1 line break indication	RUN, STARTUP, or SHUTDOWN
<b>ICS Train Isolation (Train 2)</b>	
ICS train 2 line break indication	RUN, STARTUP, or SHUTDOWN
<b>Diverse ICS Train Isolation (Train 3)</b>	
ICS train 3 line break indication	RUN, STARTUP, or SHUTDOWN

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 7.3-2: BWRX-300 Candidate Accident Monitoring Variables**

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
None	A	BWX-300 has no specific planned manually controlled actions for safety systems to perform their safety functions	Not Applicable (NA)	NA	NA	NA	NA
Containment Sump Level	B, C	Boiling Water Reactor (BWR) legacy knowledge	Assess maintenance of reactor coolant pressure boundary function. Indicate potential breach of reactor coolant pressure boundary	< 8 hours after an event involving steam, or water may be discharging into the containment	Y	Y	Designated for monitoring fundamental safety functions during the implementation of emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs) and shall comply with the requirements for safety classifications
Neutron Flux	B	BWR legacy knowledge	Assess shutdown function	< 72 hours after a scram	N	N	NEDO-31558, BWR Owners Group Proposed Neutron Monitoring System Post-Accident Monitoring Functional Criteria
Reactor Pressure	B, C, D	BWR legacy knowledge	Assess maintenance of Reactor Coolant Pressure Boundary (RCPB) function Indicate potential breach of RCPB Confirm ICS operation and performance to accomplish cooldown to stable shutdown state	< 72 hours after an event initiation	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
Reactor Pressure Vessel Dome Pressure	B	BWR legacy knowledge	Pressure Control Assess RCPB integrity	< 72 hours after an event initiation	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications
Reactor Isolation Valve Positions	B	BWR legacy knowledge	Confirm accomplishment of RPV isolation	< 1 hour after isolation actuation demand	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications
RPV Water Level	B	BWR legacy knowledge	Level Control Assess maintaining adequate core cooling by submergence Indicate potential breach of fuel clad	< 72 hours after an event initiation	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications
Containment Area Radiation Level	C, E	BWR legacy knowledge	Indicate potential for a breach of fission product barriers (fuel clad, RCPB) Indicate actual breach of fission product barriers (fuel clad, RCPB) Monitor radiation levels and radioactivity in the plant environs	< 72 hours after an event involving elevated radiation	Y	N	
Containment Hydrogen Concentration	C, F	BWR legacy knowledge	Indicate an actual breach of fission product barriers (fuel clad) Indicate potential for a breach of fission product barriers (fuel clad)	< 4 hours after placing monitor in service	Y	N	Provided for severe accident mitigation so commercial grade equipment is acceptable

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
Containment Oxygen Concentration	C, F	BWR legacy knowledge	Indicate an actual breach of fission product barriers (containment) Indicate potential for a breach of fission product barriers (containment)	< 4 hours after placing monitor in service	Y	N	Provided for severe accident mitigation so commercial grade equipment is acceptable
Containment Pressure	C, D	BWR legacy knowledge	Indicate potential breach of reactor coolant pressure boundary. Indicate potential breach of containment boundary. Indicate performance of Primary Containment System	< 72 hours after an event involving elevated containment pressure	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications
Isolation Condenser Condensate Line Flowrate	C, D	BWR legacy knowledge	Indicate potential for a breach of fission product barriers (RCPB) Indicate actual breach of fission product barriers (fuel clad, RCPB) Indicate performance of ICS safety system	< 72 hours after initiation	Y	Y	
Isolation Condenser Steam Line Flowrate	C, D	BWR legacy knowledge	Indicate potential for a breach of fission product barriers (RCPB) Indicate actual breach of fission product barriers (fuel clad, RCPB) Indicate performance of ICS safety system	< 72 hours after initiation	Y	Y	

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
Off Gas Activity	C	BWR legacy knowledge	Indicate potential breach of fuel clad Monitoring performed by normal operating system	< 8 hours after a radioactive release event	Y	Y	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and shall comply with the requirements for safety classifications
Boron Injection System Indications	D	BWR legacy knowledge	Indicate performance of required auxiliary support features	< 2 hours after initiation	N	N	Not associated with any events requiring environmental or seismic qualification
Containment Isolation Valve Positions	D	BWR legacy knowledge	Verify containment system status	< 1 hour after Isolation actuation demand	Y	Y	Primary accident monitoring information for the containment is provided by reactor pressure, reactor water level, and containment pressure
Containment Temperature	D	BWR legacy knowledge	Indicate performance of Primary Containment System	< 8 hours after an event involving elevated containment temperature	Y	Y	

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
Control Rod Position	D	BWR legacy knowledge	Indicate performance of other systems	< 1 hour after a Scram	N	N	The rod position indication is a normal operating system that is not required to be seismically designed. Its function is completed before experiencing a harsh environment. Also, the proper functioning of the Hydraulic Scram and Control Rod Drives can be inferred from other parameters
Electrical Power Status	D	BWR legacy knowledge	Indicate performance of required auxiliary support features	< 72 hours after a loss of an electrical power event	N	N	
Feedwater Flow	D	BWR legacy knowledge	Indicate performance of other systems	< 1 hour after a loss of feedwater flow	N	N	Not associated with any events requiring environmental or seismic qualification
Isolation Condenser Valve Positions	D	BWR legacy knowledge	Indicate performance of ICS	< 1 hour after initiation	Y	Y	
Isolation Condenser System Pool Water Level	D	BWR legacy knowledge	Indicate performance of ICS	< 72 hours after initiation	Y	Y	
Area Radiation Level	E	BWR legacy knowledge	Monitor radiation levels and radioactivity in the plant environs	< 72 hours after an event involving elevated radiation	N	N	



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Variable	Type	Selection Reason	Critical Safety Functions to Monitor	Estimated Mission Time	EQ	SQ	Remarks
Control Building Air Intake HVAC Radiation Level	E	BWR legacy knowledge	Monitor radiation levels and radioactivity in the plant environs Monitor radiation and radioactivity levels in the control room and selected plant areas where access may be required for plant recovery	< 1 hour after an event of detecting elevated radiation in the control room	N	N	
Effluent radioactivity - noble gases	E	BWR legacy knowledge	Monitor the magnitude of releases of radioactive materials through identified pathways	< 8 hours after a radioactive release event	N	N	
Meteorological Data (Wind Speed, Wind Direction, and Atmospheric Stability)	E	BWR legacy knowledge	Monitor the environmental conditions used to determine the impact of releases of radioactive materials through identified pathways	< 72 hours after a Site Area Emergency is declared	N	N	
Plant Environment Radiation/Radioactivity Levels	E	BWR legacy knowledge	Monitor the magnitude of releases of radioactive materials through identified pathways	< 8 hours after a radioactive release event	N	N	
Spent Fuel Pool Level	F	IEC 63147	Indicate spent fuel pool fuel damage under severe accident conditions	< 72 hours after an event initiation	Y	Y	Provided for severe accident mitigation so commercial grade equipment is acceptable

(1) EQ = Environmental Qualification, SQ = Seismic Qualification

(2) Estimated Mission Time are provided to support and inform early project phase work and design decisions related to post-accident monitoring and applicable support systems.

**Table 7.3-3: DL4a Diverse Protection System Functions and Initiating Signals**

Initiating Signal	Reactor Mode Switch Position
<b>Diverse Hydraulic Scram</b>	
HIGH RPV pressure	RUN or STARTUP
LOW reactor water level [L3]	RUN or STARTUP
HIGH containment pressure	RUN or STARTUP
HIGH main condenser pressure	RUN
Line break indication (MSL, FW, or ICS)	RUN, STARTUP, or SHUTDOWN
Manual initiation	RUN or STARTUP
<b>CRD Fast Motor Run-In</b>	
High Flux After Scram	RUN or STARTUP
<b>Isolation Condenser Actuation</b>	
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
IC Line Break Indication (unaffected ICS trains initiated)	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a MSL line break	RUN, STARTUP, or SHUTDOWN
Manual initiation	RUN, STARTUP, or SHUTDOWN
<b>RPV and Containment Isolation</b>	
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
<b>MSRIV and MSCIV Isolation</b>	
Sustained LOW feedwater flow	RUN, STARTUP, or SHUTDOWN
MSL break indication	RUN, STARTUP, or SHUTDOWN
FW line break indication	RUN, STARTUP, or SHUTDOWN
SDC line break indication	RUN, STARTUP, or SHUTDOWN
<b>Feedwater Isolation</b>	
HIGH RPV level [L9]	RUN
Loss of normal FW flow	RUN, STARTUP, or SHUTDOWN
<b>Feedwater and Condensate Pump Trip</b>	
Detection of FW line Break	RUN, STARTUP, or SHUTDOWN
High Flux after Scram	RUN or STARTUP

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

Initiating Signal	Reactor Mode Switch Position
<b>Feedwater and SDC Isolation</b>	
Indication of SDC line break	RUN, STARTUP, or SHUTDOWN
Detection of FW line Break	RUN, STARTUP, or SHUTDOWN
<b>CUW Isolation</b>	
CUW line break indication	RUN, STARTUP, or SHUTDOWN
<b>Cavity Pool Makeup from ICS Pools</b>	
RPV level reduction	REFUEL
<b>ICS Train Isolation</b>	
Manual (Per Train)	RUN, STARTUP, or SHUTDOWN

**Table 7.3-4: DL2 Anticipatory Trip and Block Functions and Initiating Signals**

Initiating Signal	Reactor Mode Switch Position
<b>Reactor Scram</b>	
Turbine Trip Demand	RUN
Closure of all MSRIV and MSCIV (<90% Open)	RUN
Short reactor period (at low power) from WRNM	STARTUP
LOW bus voltage (from R20 electrical system)	RUN or STARTUP
HIGH main condenser pressure	RUN
High RPV level	RUN or STARTUP
Sustained LOW feedwater flow	RUN
Manual initiation	RUN or STARTUP
<b>Control Rod Withdrawal Block</b>	
ATLM	RUN
MRBM	RUN
Short Reactor Period	STARTUP
Control Rod Separation	RUN, STARTUP, or SHUTDOWN

**Table 7.3-5: C10 SC1 I&C Compliance Alignment**

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Qualified I&C	Section 7.5, Design Rules, and Limits (with Reliable I&C) Section 7.8, Equipment Environmental Qualification Section 7.9.2, Use of Computer-Based Systems or Equipment Section 7.13, Seismic Qualification and Design
Reliable I&C	Section 7.5, Design Rules, and Limits (with Qualified I&C) Section 7.6, Design for Reliability (with Robust I&C) Section 7.9.1, Instrumentation and Control – General (with Robust I&C)
Robust I&C	Section 7.6, Design for Reliability (with Reliable I&C) Section 7.9.1, Instrumentation and Control – General (with Reliable I&C) Section 7.21, Human Factors (partial with Chapter 18)
Secure I&C	Section 7.22.4, Cyber Security (partial with Security Annex)
Defence-In-Depth	Section 4.3.1, Defence-in-Depth (with Robust I&C) Section 7.6.1, Common Cause Failures (partial with Robust I&C)
Operator Interface	Section 7.9.3, Accident Monitoring Instrumentation Section 7.21, Human Factors (partial with Chapter 18) Section 8.10.1, Main Control Room (partial with Chapter 18) Section 8.10.2, Secondary Control Room (partial with Chapter 18) Section 8.10.4, Credit for Operator Action (partial with Chapter 18)
I&C Performance Objectives	Section 9, Safety Analysis (partial with Chapter 15)
<b>I&amp;C Topic</b>	<b>IEC 61513 Section</b>
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defence-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
<b>I&amp;C Topic</b>	<b>IEEE Std 603 Clause</b>
Qualified I&C	Clauses 5.3 and 5.4
Reliable I&C	Clauses 5.10, 5.11, 5.12, 5.14, 5.15, 6.5, 6.7, 7.5, and 8.3
Robust I&C	Clauses 5.1, 5.2, 5.5 - 5.8, 5.13, 6.1, 6.3, 6.6, 7.1, 7.3, and 7.4
Secure I&C	Clause 5.9
Defence-In-Depth	Clauses 6.2 and 7.2

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Operator Interface	Clauses 5.8, 5.14, 6.2 and 7.2
I&C Performance Objectives	Clauses 4, 6.4, and 6.8

**Table 7.3-6: C20 Defense Line 4a Compliance Alignment**

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Qualified I&C	Section 7.5, Design Rules, and Limits (with Reliable I&C) Section 7.8, Equipment Environmental Qualification Section 7.9.2, Use of Computer-Based Systems or Equipment Section 7.13, Seismic Qualification and Design
Reliable I&C	Section 7.5, Design Rules, and Limits (with Qualified I&C) Section 7.6, Design for Reliability (with Robust I&C) Section 7.9.1, Instrumentation and Control – General (with Robust I&C)
Robust I&C	Section 7.6, Design for Reliability (with Reliable I&C) Section 7.9.1, Instrumentation and Control – General (with Reliable I&C)
Secure I&C	Section 7.22.4, Cyber Security (partial with Security Annex)
Defence-In-Depth	Section 4.3.1, Defence-in-Depth (with Robust I&C) Section 7.6.1, Common Cause Failures (with Robust I&C)
Operator Interface	Section 7.21, Human Factors (partial with Chapter 18) Section 8.10.1, Main Control Room (partial with Chapter 18) Section 8.10.2, Secondary Control Room (partial with Chapter 18) Section 8.10.4, Credit for Operator Action (partial with Chapter 18)
I&C Performance Objectives	Section 9, Safety Analysis (partial with Chapter 15)
<b>I&amp;C Topic</b>	<b>IEC 61513 Section</b>
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defence-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
<b>I&amp;C Topic</b>	<b>IEEE Std 603 Clause</b>
Robust I&C	Clauses 5.6 and 6.3
Defence-In-Depth	Clauses 5.1, 6.2 and 7.2
I&C Performance Objectives	Clauses 4, 6.4, and 6.8

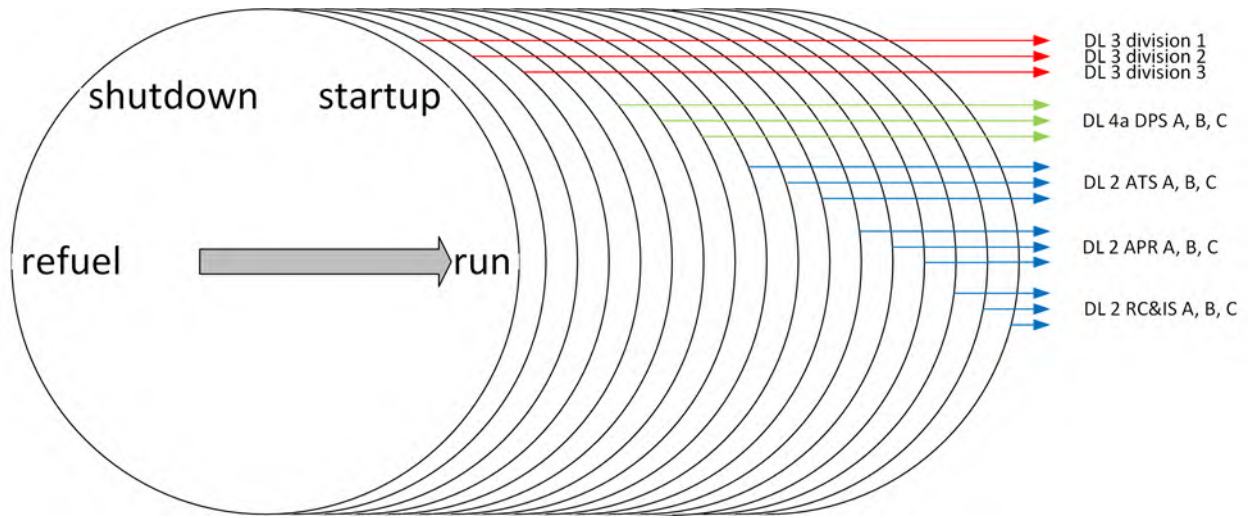
**Table 7.3-7: C20 Defense Line 2 Nuclear Segment Compliance Alignment**

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Qualified I&C	Section 75.5, Design Rules, and Limits (with Reliable I&C) Section 7.8, Equipment Environmental Qualification Section 7.9.2, Use of Computer-Based Systems or Equipment Section 7.13, Seismic Qualification and Design
Reliable I&C	Section 7.5, Design Rules, and Limits (with Qualified I&C) Section 7.6, Design for Reliability (with Robust I&C) Section 7.9.1, Instrumentation and Control – General (with Robust I&C)
Robust I&C	Section 7.6, Design for Reliability (with Reliable I&C) Section 7.9.1, Instrumentation and Control – General (with Reliable I&C)
Secure I&C	Section 7.22.4, Cyber Security (partial with Security Annex)
Defence-In-Depth	Section 4.3.1, Defence-in-Depth (with Robust I&C) Section 7.6.1, Common Cause Failures (with Robust I&C)
Operator Interface	Section 7.9.3, Accident Monitoring Instrumentation Section 7.21, Human Factors (partial with Chapter 18) Section 8.10.1, Main Control Room (partial with Chapter 18) Section 8.10.1.1, Safety Parameter Display System Section 8.10.2, Secondary Control Room (partial with Chapter 18) Section 8.10.4, Credit for Operator Action (partial with Chapter 18)
I&C Performance Objectives	Section 9, Safety Analysis (partial with Chapter 15)
<b>I&amp;C Topic</b>	<b>IEC 61513 Section</b>
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defence-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
<b>I&amp;C Topic</b>	<b>IEEE Std 603 Clause</b>
Robust I&C	Clauses 5.6 and 6.3
I&C Performance Objectives	Clauses 4, 6.4, and 6.8



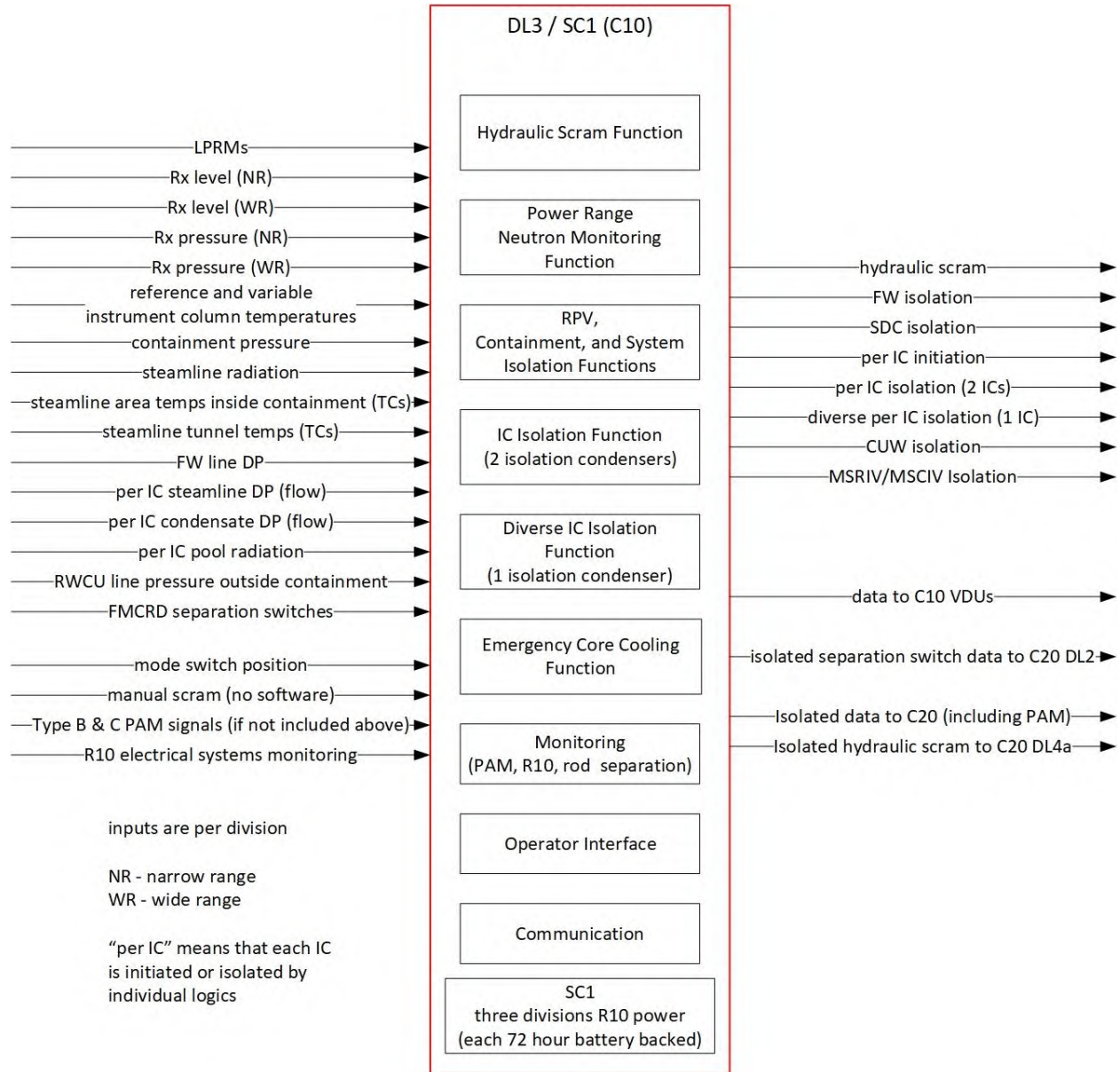
**Table 7.3-8: C30 BOP Segment Compliance Alignment**

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Qualified I&C	Section 7.5, Design Rules and Limits (with Reliable I&C) Section 7.8, Equipment Environmental Qualification Section 7.9.2, Use of Computer-Based Systems or Equipment Section 7.13, Seismic Qualification and Design
Reliable I&C	Section 7.5, Design Rules and Limits (with Qualified I&C) Section 7.6, Design for Reliability (with Robust I&C) Section 7.9.1, Instrumentation and Control – General (with Robust I&C)
Robust I&C	Section 7.6, Design for Reliability (with Reliable I&C) Section 7.9.1, Instrumentation and Control – General (with Reliable I&C)
Secure I&C	Section 7.22.4, Cyber Security (partial with Security Annex)
Defence-In-Depth	Section 4.3.1, Defence-in-Depth (with Robust I&C)
Operator Interface	Section 7.21, Human Factors (partial with Chapter 18) Section 8.10.1, Main Control Room (partial with Chapter 18)
I&C Performance Objectives	Section 9, Safety Analysis (partial with Chapter 15)



**Figure 7.3-1: Reactor Mode Switch**

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.3-2: DL3 Safety Class 1 Functional Architecture**

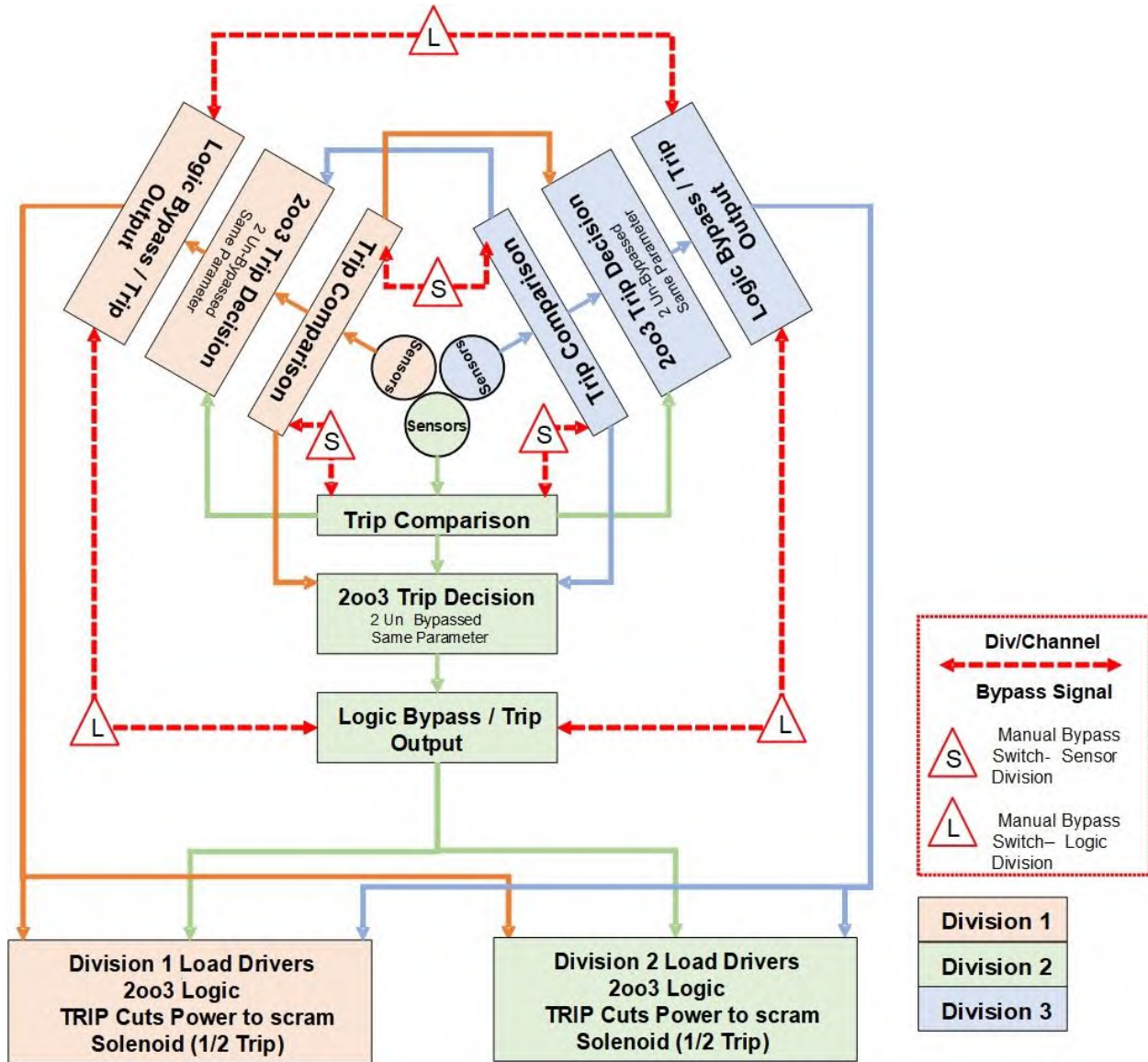
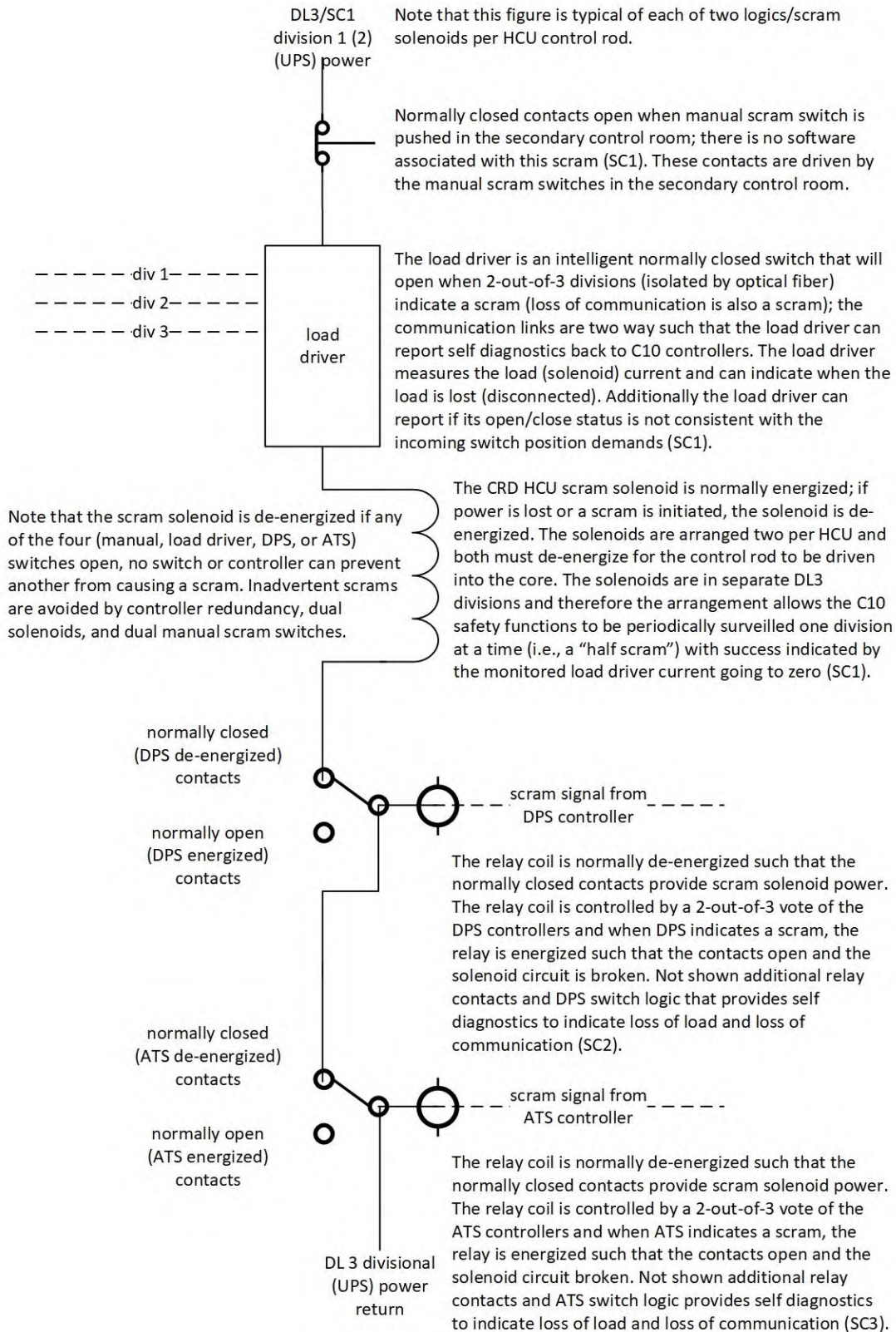


Figure 7.3-3: BWRX-300 DL3 Actuation Logic

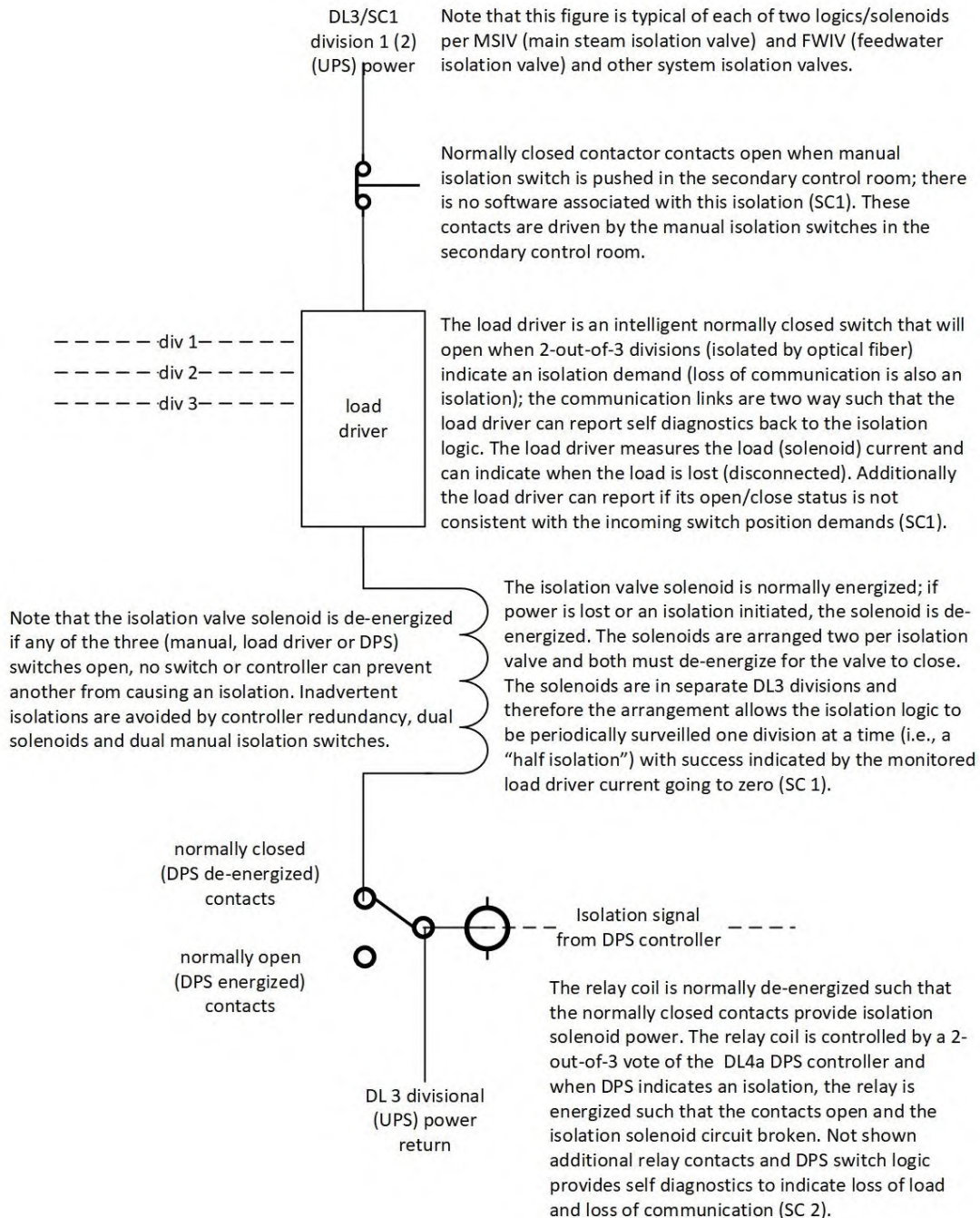


NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.3-4: BWRX-300 Hydraulic Scram Solenoid Control and Power**

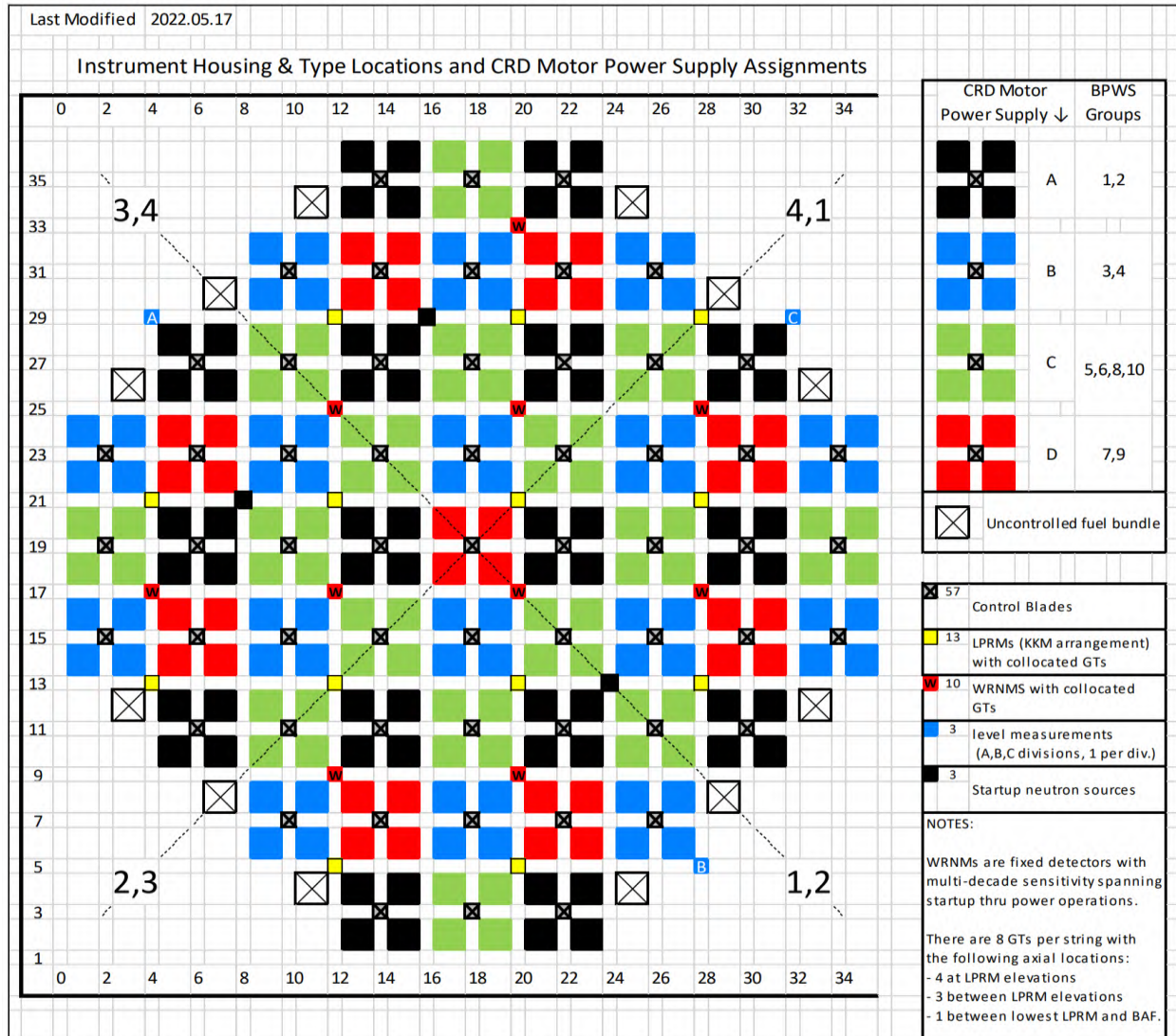
NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.3-5: BWRX-300 Isolation Valve Solenoid Control and Power**



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

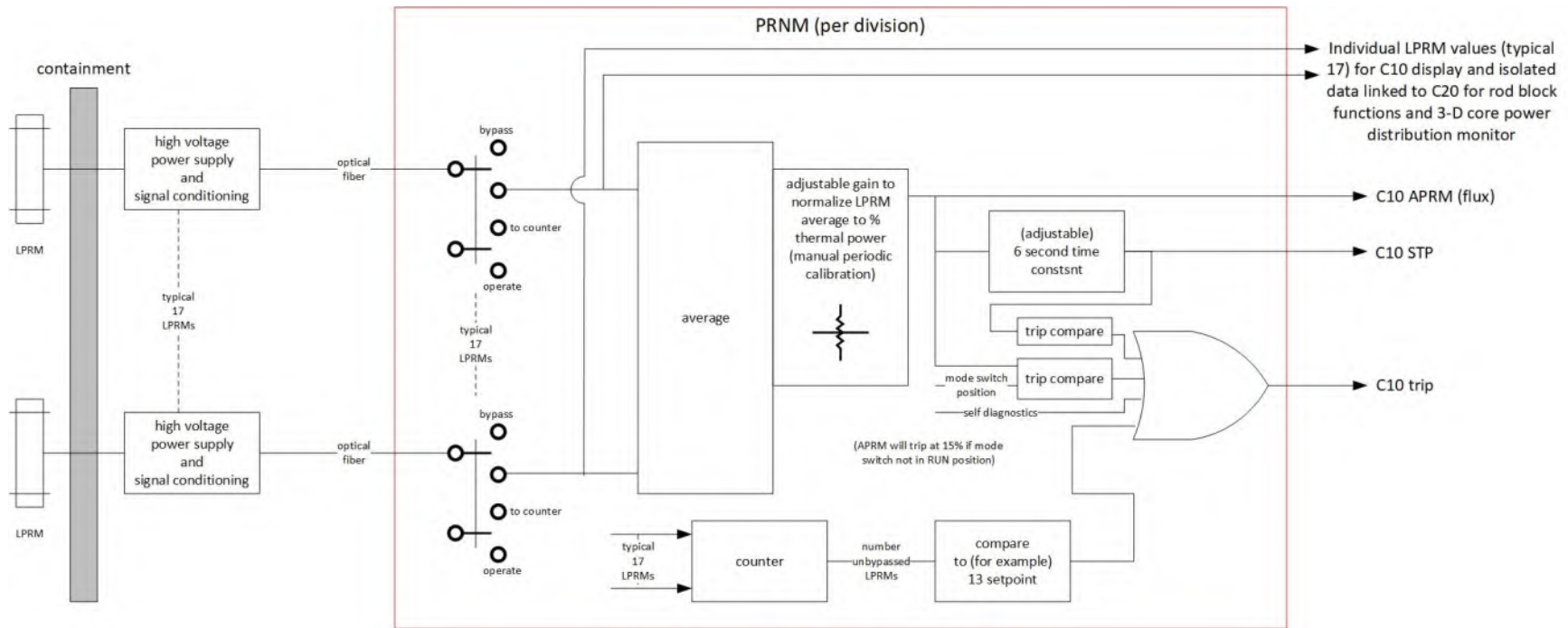


Figure 7.3-7: DL3 Power Range Neutron Monitoring Block Diagram



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

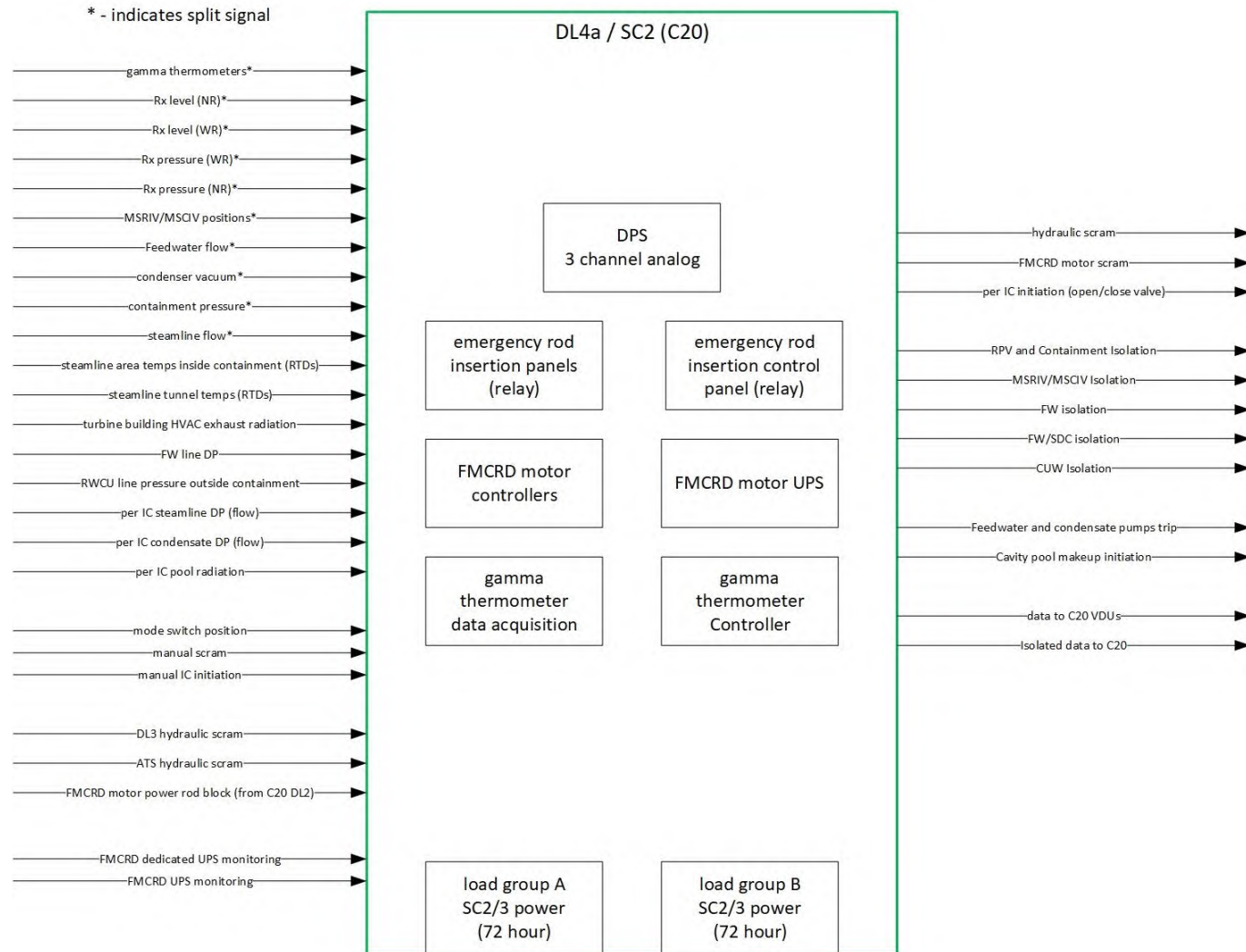


Figure 7.3-8: DL4a Safety Class 2 Functional Architecture

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

splitters are SC2 classified

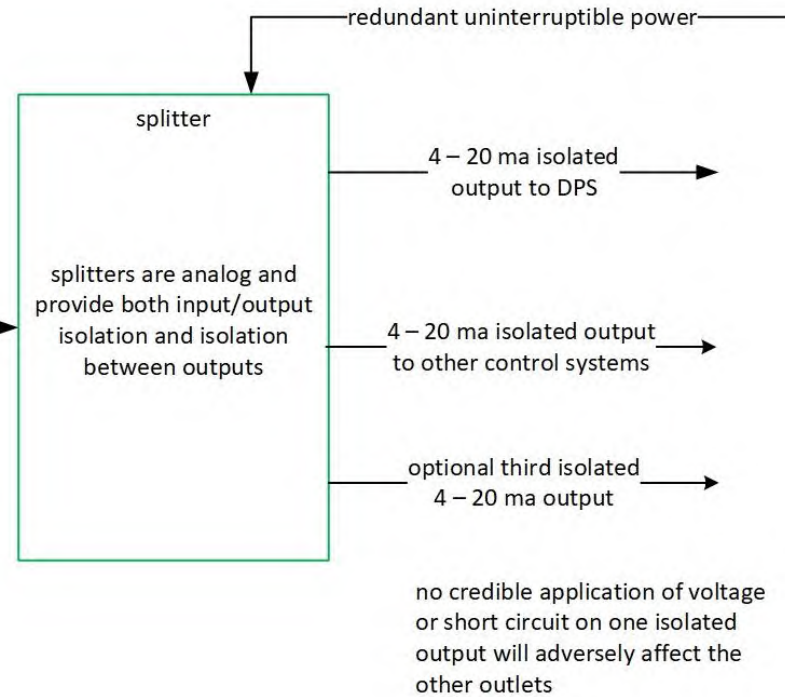
splitters are completely analog and  
do not use software

splitters are per signal and are  
arranged such that a splitter failure  
will not adversely affect other,  
triplicated signals

typical 4 – 20 ma signal

typical signals are directly from a  
transmitter like level, pressure or  
flow

other signal types like RTDs require  
signal conditioning to convert  
them to 4 – 20 ma signals before  
the splitter



**Figure 7.3-9: DL4a Analog Signal Splitters**

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

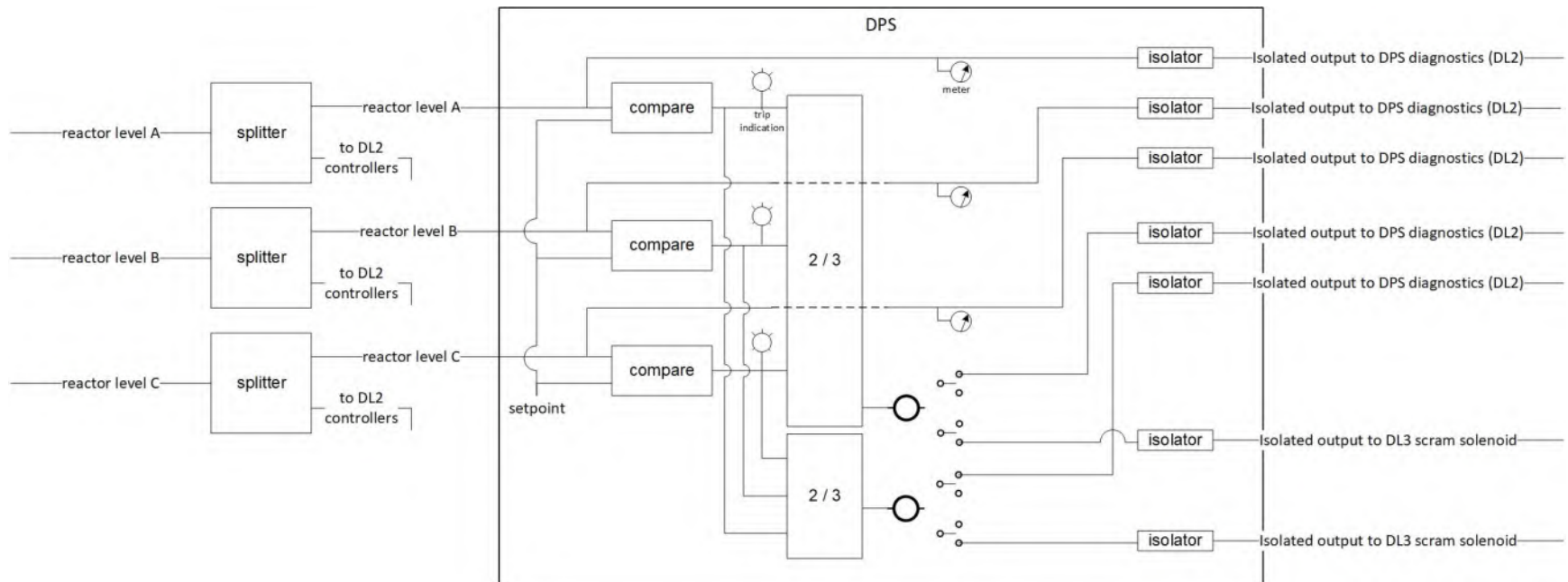
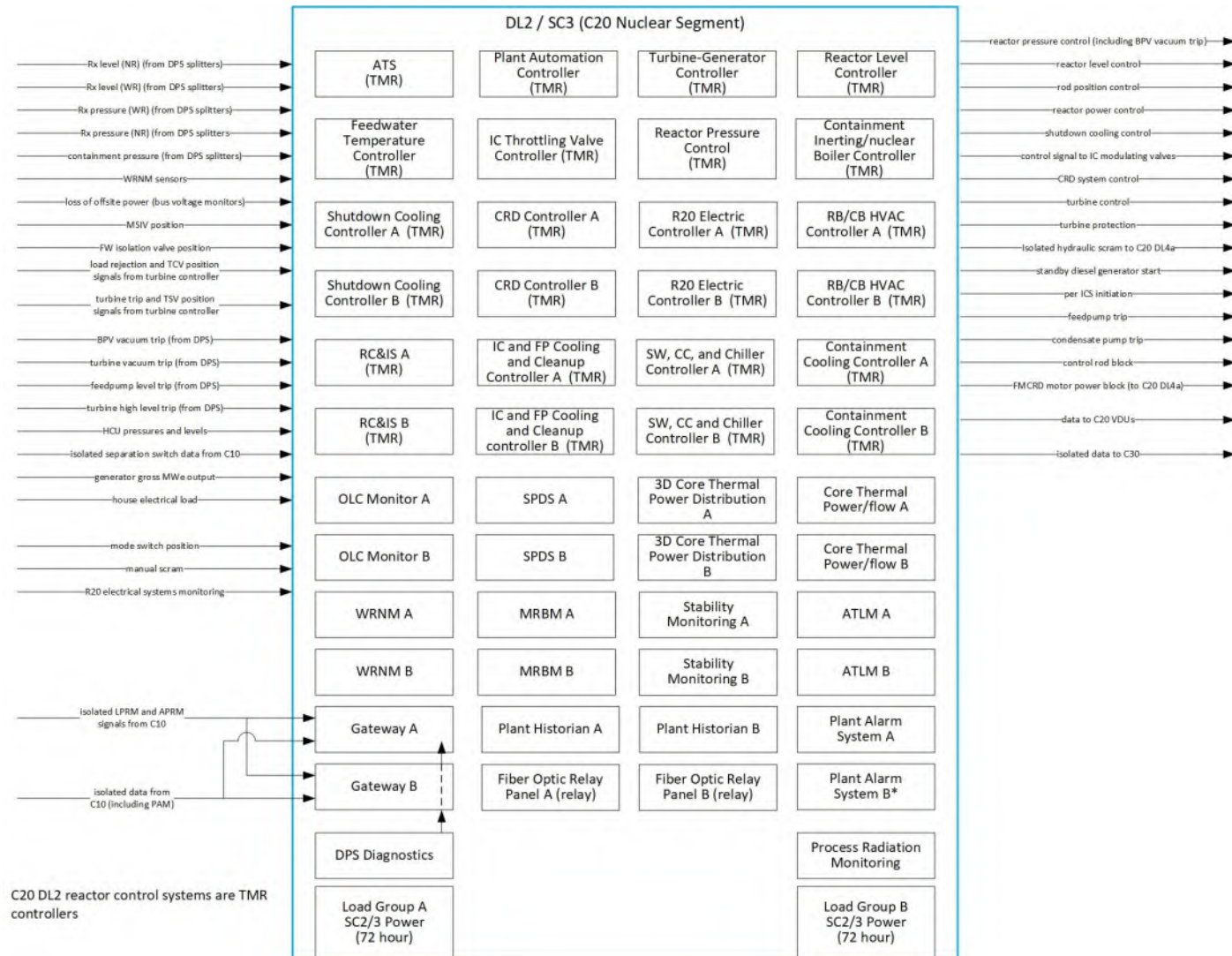


Figure 7.3-10: Representative Diverse Protection System Scram Block Diagram

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.3-11: DL2 Safety Class 3 Nuclear Segment Functional Architecture**

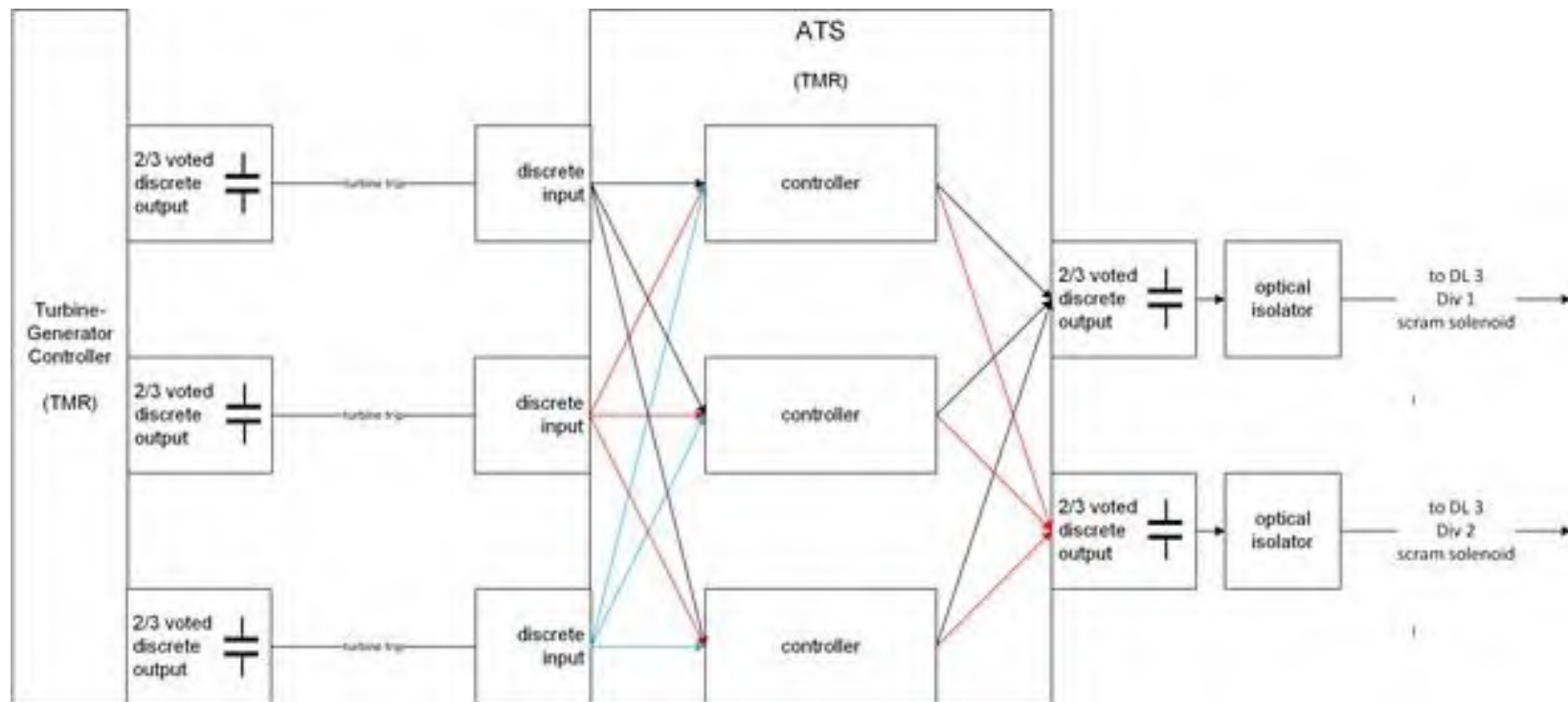


Figure 7.3-12: Example Anticipatory Trip System Scram Block Diagram



NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION

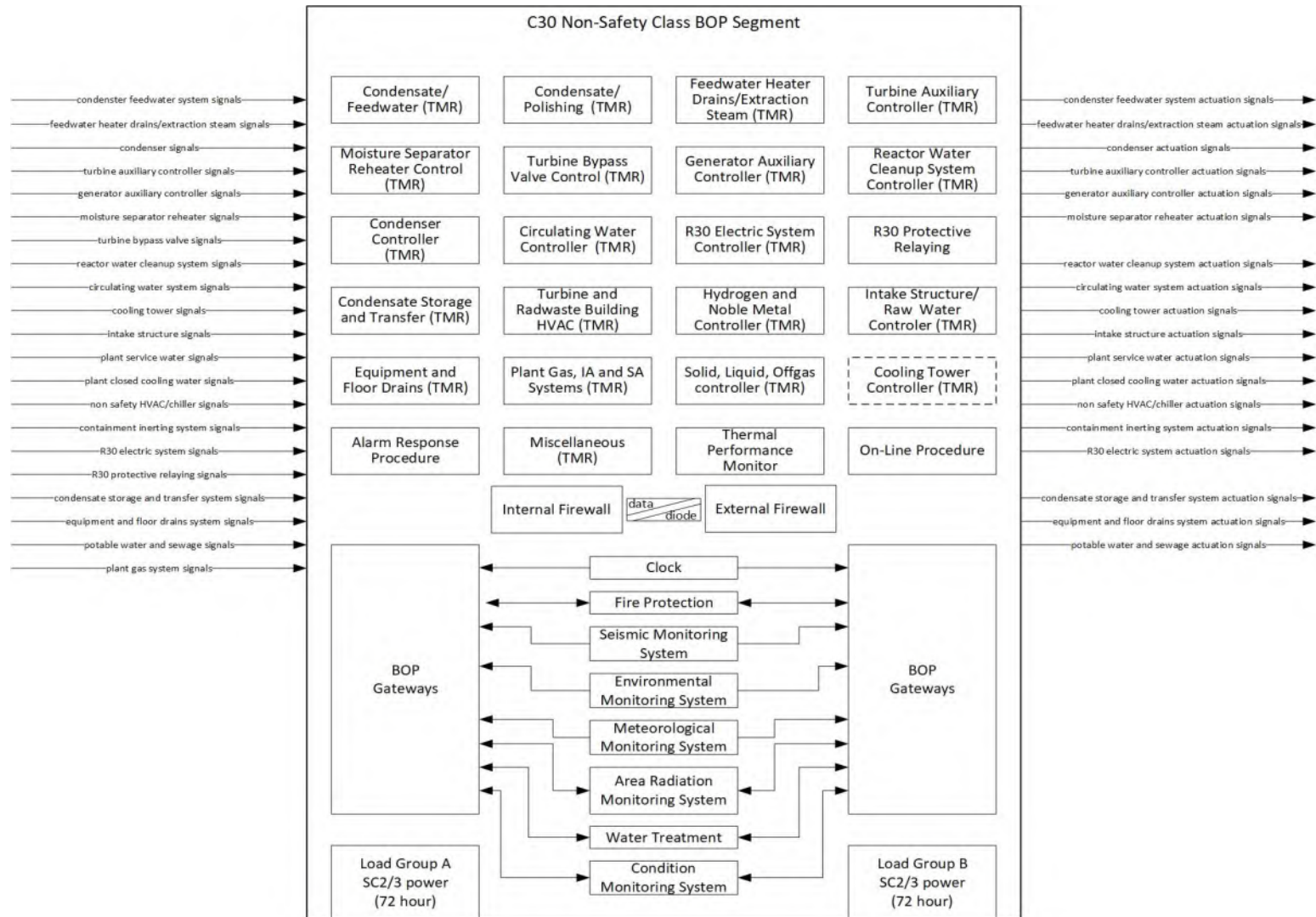


Figure 7.3-13: C30 Non-Safety Class BOP Functional Architecture

## **7.4 Digital Instrumentation and Control System Development Processes**

The BWRX-300 I&C systems design is the result of a past and continuing top – down systematic design process. This process makes risk-informed decisions about the scope and level of design detail required at various stages of the BWRX-300 design process. The BWRX-300 I&C design process is derived from IEC 61513 (Reference 7.4-1). The BWRX-300 I&C design process is informed by first-hand experience and lessons learned from already licensed and delivered BWR I&C designs and proposed new plant I&C designs (e.g., U.S., Finland, Taiwan, and United Kingdom).

### **7.4.1 Design Control**

The BWRX-300 I&C design process is used to ensure that the design and its associated design documentation meet applicable technical requirements, regulatory requirements, codes and standards, and contractual requirements. It is an iterative and recursive design process to allow sequencing and development of requirements. The design process has four design phases (i.e., Requirements, Conceptual, Preliminary, Detailed). The design phases are defined by the maturity of each of the requirement levels, and deliverables are defined for each phase.

1. **Requirements Design Phase:** The requirements design phase focuses on maturity of the requirements of the design using an abbreviated concept design cycle to more closely examine the design and technology selections suited for demonstration on the desired timeline. This phase focuses on rapid study and design of requirements, systems, and component design alternatives to support timely Licence to Construction submittal, and refinement of the selections into an optimized safety and licensing package.
2. **Conceptual Design Phase:** The conceptual design phase focuses on developing the specific design definition of systems and components through component and system level design analysis. This includes detailed system and component analysis, specification and drawing development, detailed manufacturing and fabrication analyses, and long lead procurement activities. Interim verification and validation work is conducted to confirm the adequacy of design requirements and objectives.
3. **Preliminary Design Phase:** The preliminary design phase focuses on the effort to prepare the design for construction, detailed component design, and effort to support equipment procurement/fabrication required at this stage of design. Key system and component design documents and calculations of record are issued to support licensing and construction, and equipment specifications are finalized for procurement. Comprehensive design verification and validation is performed to support submittal of the licence to operate.
4. **Detailed Design Phase:** The detailed design phase focuses on finalization and verification of remaining system and component design in preparation for the complete scope of construction activities. The remaining equipment specifications are finalized and executed for procurements, and remaining system and component design documents and calculations of record are issued.

### **7.4.2 Defence-in-Depth and Architecture Design**

A structured, formal approach to defining the overall I&C architecture is a prerequisite for successful licensing of a new nuclear plant design. The development process output includes a definition of each I&C system in terms of its assigned functions, its safety classification, and its relationships to other systems. The design of the I&C architecture provides a top-level definition of the I&C systems of the BWRX-300 and the communication between BWRX-300 systems. The design process and supporting tools ensure a consistent interface between these systems. The BWRX-300 I&C Architecture Design Process is shown in Figure 7.4-1.

#### **7.4.2.1 Requirements Phase and Activities**

The requirements which define the necessary functions to control, operate or monitor are specified. An I&C functional requirement is defined in such a way that it:

- Gives a complete representation of a functional objective
- Can be categorized according to its degree of importance to safety

Inputs to the I&C Functional Requirements include:

- Process and electrical system design descriptions
- Transient and accident analyses
- Human factors analyses

The requirements which define the I&C systems are required to be independent and diverse from each other are specified. An I&C architectural requirement is defined in such a way that it specifies a safety classification and level of redundancy for each I&C system.

Inputs to the Architectural Requirements include:

- Transient and accident analyses
- Severe accident analyses
- D-in-D concept for the plant
- Canadian and international regulations, guidance, and endorsed standards
- Human factors analyses

These are the constraints placed on the BWRX-300 I&C Architecture from external influences. Examples of external influences include:

- Plant building and room layouts
- Process system interfaces
- Support system designs
- Environmental conditions
- Sources of internal and external hazards
- Concept for online maintenance
- Human factors constraints
- Cyber security constraints

#### **7.4.2.2 Design Phase and Activities**

The allocation of functions activity involves the definition of a set of criteria to govern the process of allocating I&C functions to I&C systems and the act of allocating the I&C functions following the criteria. This can be done initially at the level of control and monitoring functions, but eventually needs to be done for each I&C function. I&C sub-functions are created during this process.

The Definition of Architecture activity is the top-level definition of I&C systems in terms of the functional scope of each I&C system. At a minimum, the definition of an I&C system comprises



its safety classification, its design basis function, and its position in the lines of defence. This work can also include top-level assignment of I&C systems to physical locations in the plant.

Both the allocation of functions and definition of architecture are iterative, both to each other and as the BWRX-300 plant design progresses.

The Definition of Interfaces activity represents the organization and documentation of the interfaces created by the architecture definition and function allocation processes. The key elements to document are 1) basis (need) for the interface, 2) top-level nature of the interface (e.g., hardware, data communication, point-to-point, and switched network), and 3) primary requirements on the interface (e.g., independence, diversity, and failure modes). Interfaces also include support systems for the I&C system.

Human Factor Interfaces are integrated in the Architecture and Design structure for each system control and monitoring areas including the MCR, supplementary control points (e.g., SCR), local control panels, and any emergency control location. The design of the I&C architecture implements the principles for plant operation established in the BWRX-300 plant design including 1) requirements between automatic signals and manually initiated control signals, 2) requirements between the different HSI systems during normal, accident, and post-accident operation, and 3) requirements between normal and backup HSI systems (including any switchover rules). The human factors engineering program is described in Chapter 18.

### **7.4.3 Instrumentation and Control System Life Cycle**

The I&C System Life Cycle applied to each I&C system follows the overall lifecycle of IEC 61513 and the System Engineering “V” model to show top-down design and bottoms up integration for verification and validation testing, as shown in Figure 7.4-2. This plan includes the project timeline in X axis and vertical level project progression with verification and validation arrows included. Configuration management requirements for the project are defined in Configuration Management Plan for the project. The Requirements Management Plan for the project provides direction for managing requirements appearing in project documents developed during execution of the design process.

Each PLICA lifecycle phase within the lifecycle contains activities and output documentation aimed at fulfilling the purpose of the phase. The phases are not strictly sequential in their execution; rather, they are iterative. For example, decisions and discoveries made in a ‘subsequent’ phase can necessitate a return to ‘previous’ phases to modify requirements, design decisions, or output documentation. In some cases, portions of different phases can be executed in parallel. In general, a ‘subsequent’ phase can be started prior to completion of the ‘previous’ phase.

The software related activities in I&C System Life Cycle are shown in Figure 7.4-3.

### **7.4.4 Cyber Security Life Cycle**

The BWRX-300 Cyber Security Program Plan is designed to meet regulatory requirements, relevant industry standards and best practices for the design, operation, and protection of the BWRX-300 I&C systems. The BWRX-300 Cyber Security Program Plan applies advanced security principles throughout the product development and deployment lifecycle of the BWRX-300 I&C systems. The objective of the program plan is to achieve a high assurance that unauthorized access to the protection, control, and adjustment systems of the BWRX-300 is prevented. The program is based on three key principles: 1) regulatory compliance, 2) integration of cyber security controls into the BWRX-300 I&C systems development process, and 3) integration of cyber security features into the BWRX-300 I&C systems.

The BWRX-300 Cyber Security Program Plan recognizes that protecting cyber essential assets is an ongoing program throughout the I&C system development and operations stages. It is designed to:

- Implement security controls to protect the necessary digital assets from cyber attacks
- Apply and maintain D-in-D protective strategies to ensure the capability to detect, respond to, and recover from cyber attacks
- Mitigate the adverse effects of cyber attacks
- Ensure functions of protected assets are not adversely impacted due to cyber attacks

The cyber security life cycle is shown in Figure 7.4-4. The cyber security architecture security boundaries are shown in Figure 7.4-5. BWRX-300 Cyber Security Program Plan classifies the DCIS systems C10, C20 and C30 equipment with functions important to nuclear safety. These systems are protected from lower classified systems by the unidirectional communication flow between Level 3 and Level 2, as shown in Figure 7.4-5 and consistent with zone requirements in CSA N290.7:21 (Reference 7.4-2). The BWRX-300 Cyber Security Program Plan adds an additional layer of protection between C10 SC1 I&C and C20 with unidirectional communication flow between Level 4 and Level 3. Data flow and communications between defence levels are managed by boundary devices that implement unidirectional flow where required to prevent a lower level from compromising a higher level.

#### **7.4.5 Compliance Alignment**

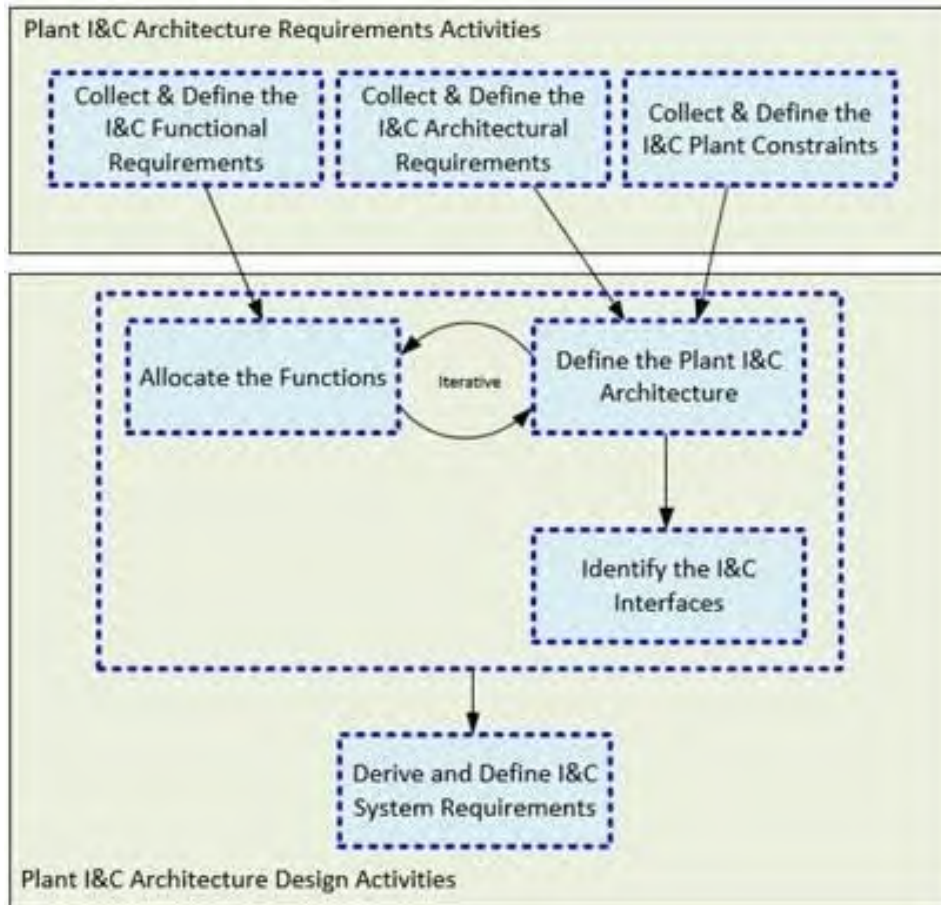
This section describes how the BWRX-300 I&C Development Life Cycle aligns to meet the regulatory guidance in CNSC REGDOC-2.5.2 and the key industry I&C standards IEC 61513 and IEEE Std 603 (Reference 7.4-3), as shown in Table 7.4-1.

#### **7.4.6 References**

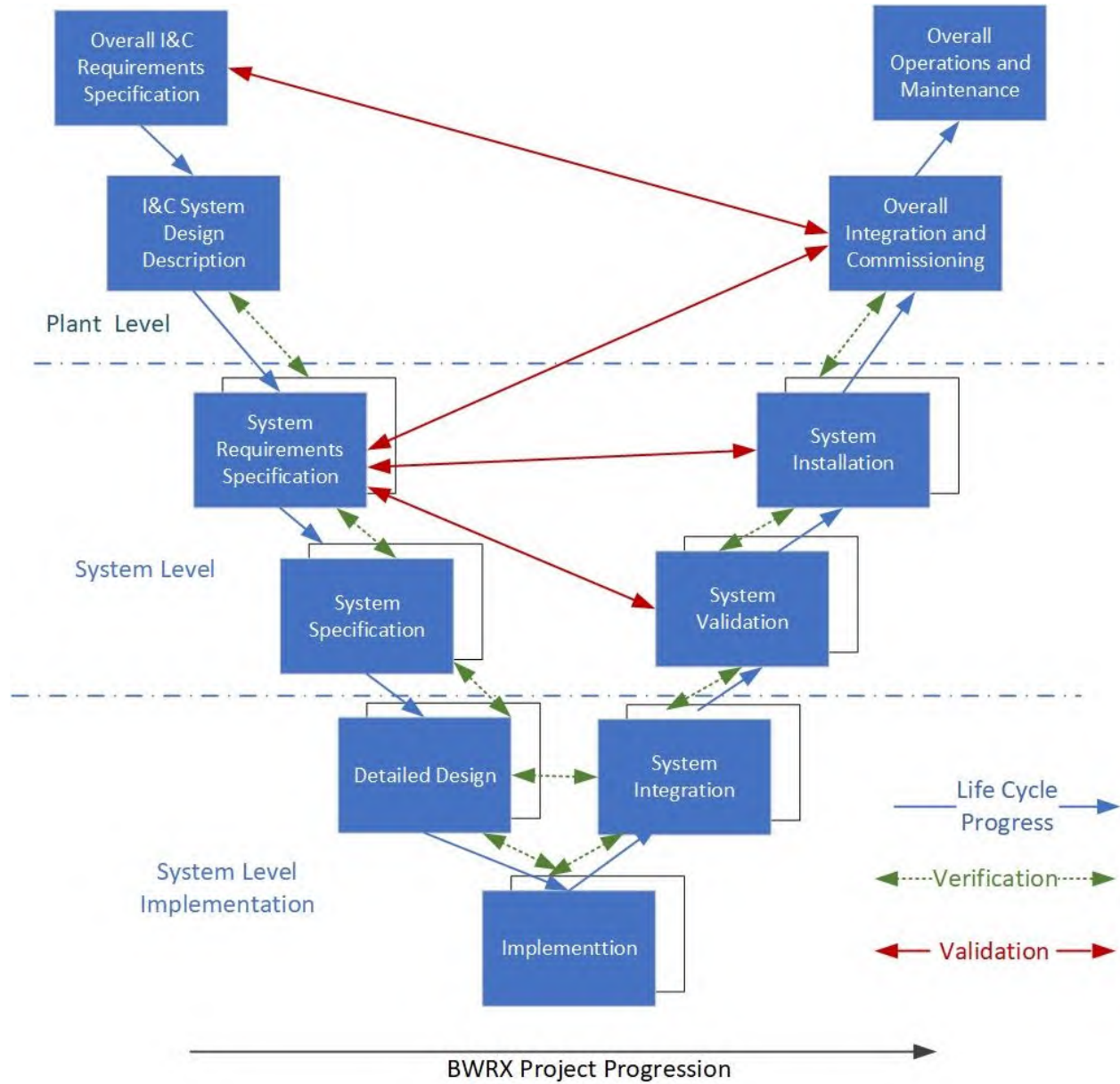
- 7.4-1 IEC 61513, "Nuclear power plants – Instrumentation and control important to safety – General requirements for systems," International Electrotechnical Commission.
- 7.4-2 CSA N290.7, "Cyber Security for Nuclear Facilities," CSA Group.
- 7.4-3 IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 7.4-4 CSA N290.14, "Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants," CSA Group.
- 7.4-5 IEC 60880, "Nuclear power plants - Instrumentation and control systems important to safety - Software aspects for computer-based systems performing category A functions," International Electrotechnical Commission.
- 7.4-6 IEC 62138, "Nuclear power plants - Instrumentation and control systems important to safety - Software aspects for computer-based systems performing category B or C functions," International Electrotechnical Commission.
- 7.4-7 IEEE 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

**Table 7.4-1: Digital I&C System Development Process Compliance Alignment**

<b>I&amp;C Topic</b>	<b>CNSC REGDOC-2.5.2 Section</b>
Qualified I&C	Section 5.3, Design Control Measures (with Chapter 17)
	Section 7.9.2, Use of Computer-Based Systems or Equipment, including CSA N290.14:15 (Reference 7.4-4)
	Section 7.22.4, Cyber Security, including CSA N290.7:21 requirements for cyber secure architecture and controls
<b>I&amp;C Topic</b>	<b>IEC 61513 Section</b>
Qualified I&C	Sections 5.5.2. and 6.2.2.7 as supplemented by IEC 60880 (Reference 7.4-5) and IEC 61238 (Reference 7.4-6)
<b>I&amp;C Topic</b>	<b>IEEE Std 603 Clause</b>
Qualified I&C	Clauses 5.3 and 5.4 as supplemented by IEEE Std 7-4.3.2 (Reference 7.4-7)



**Figure 7.4-1: BWRX-300 I&C Architecture Design Process**



**Figure 7.4-2: Overall BWRX-300 I&C System Life Cycle**

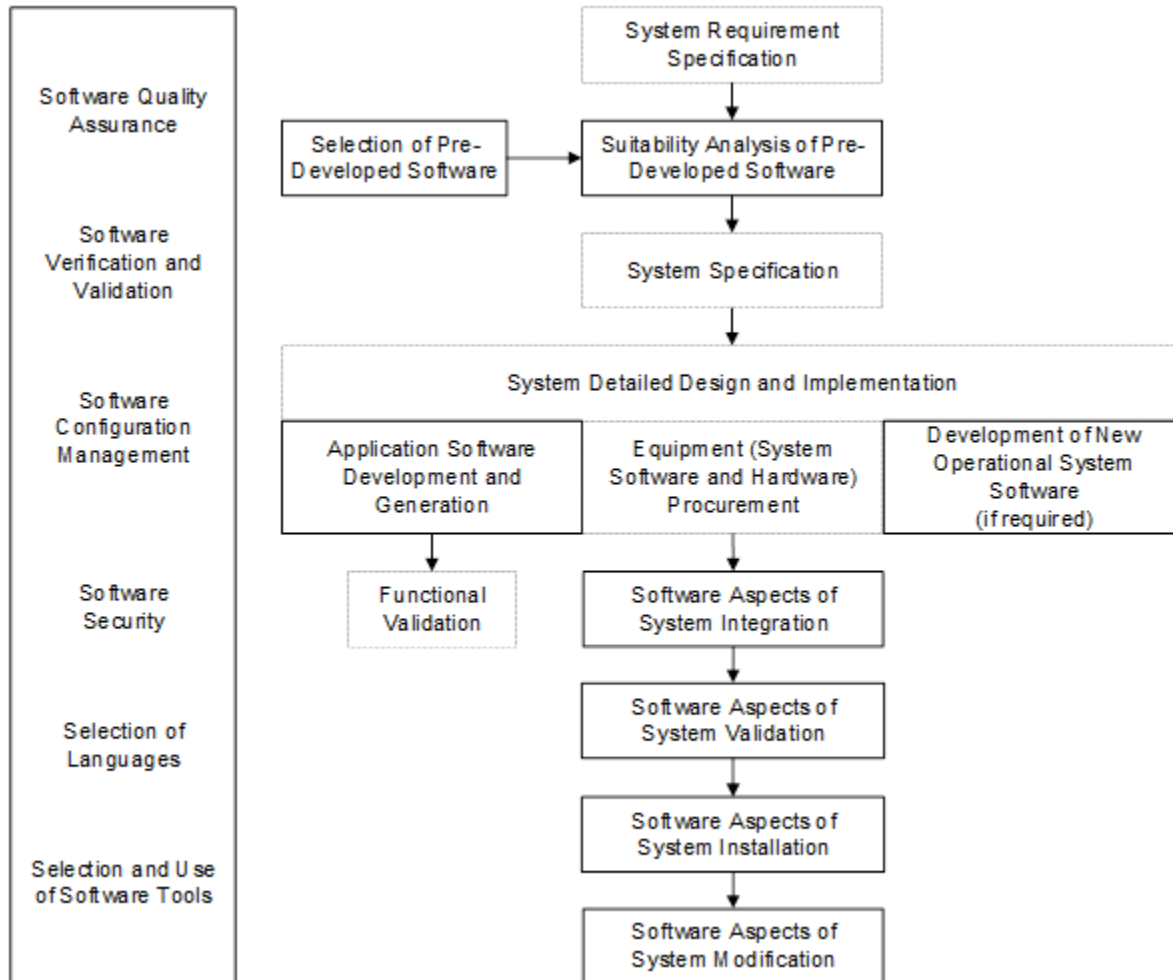
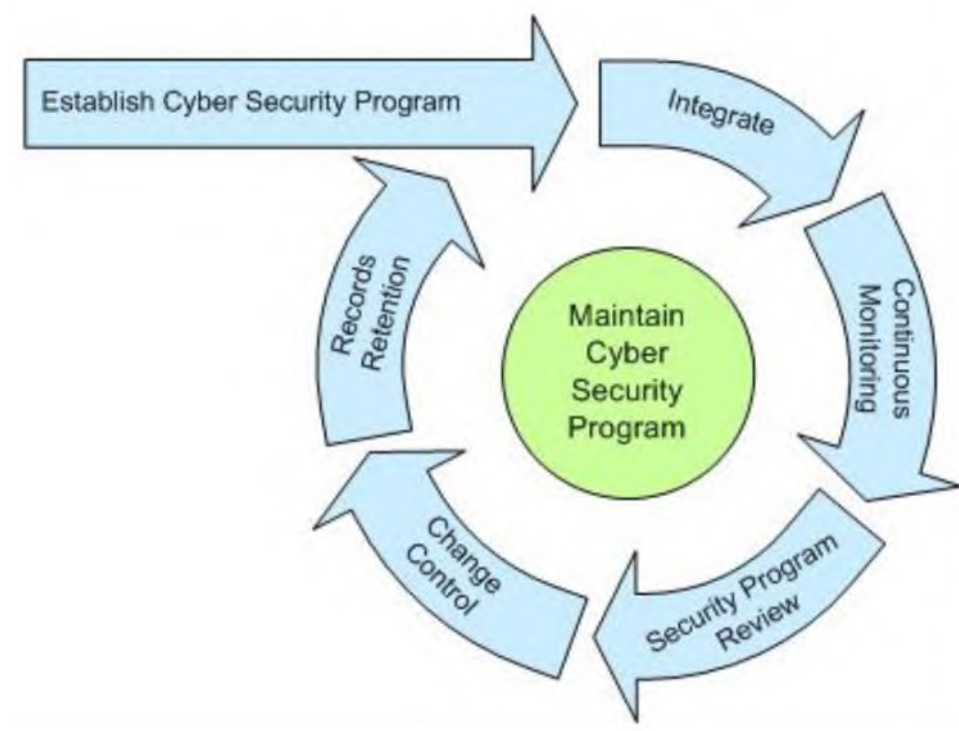


Figure 7.4-3: Software Related Activities in I&C System Life Cycle



**Figure 7.4-4: Cyber Security Life Cycle**

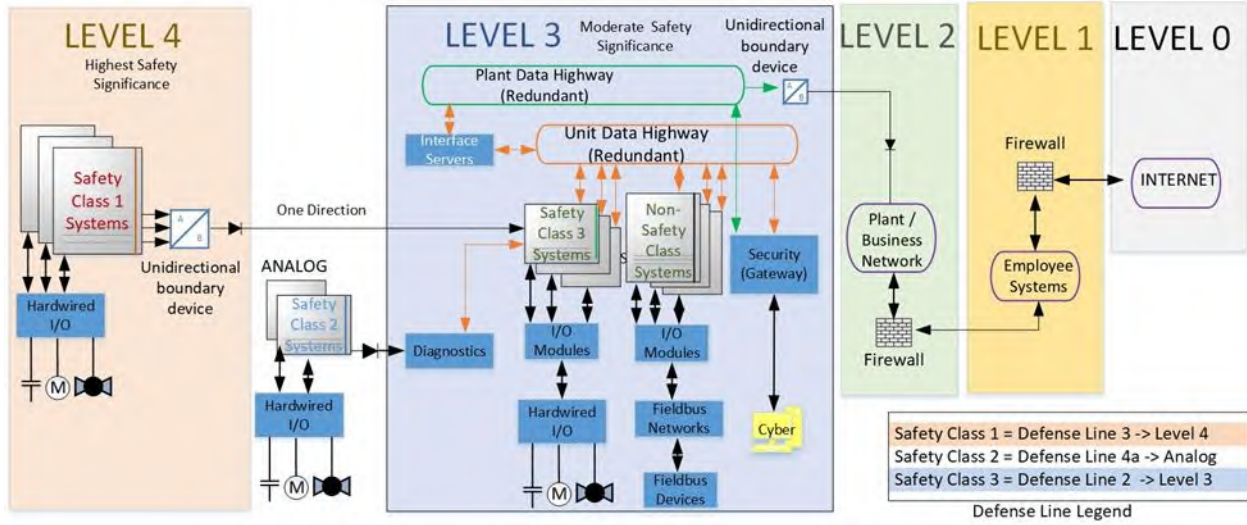


Figure 7.4-5: Cyber Security Architecture Security Boundaries



## **7.5 Instrumentation and Control in the Main Control Room**

The MCR is located in the Control Building. The MCR is the primary location for plant monitoring and control during normal, abnormal, and emergency conditions. The MCR includes controls, indications and alarms that enable operators to perform the defined set of functions during normal operation modes and PIE conditions. It is expected that the safety and HFE analyses being performed as an integral part of the design of the BWRX-300 will dictate a requirement for an alternate control location; as such, the current concept design includes the SCR. In the event the MCR becomes uninhabitable, or is expected to become uninhabitable, or if functionality is unacceptably impaired, personnel will move to the SCR. The intended functionality of the SCR is to maintain the unit in a safe shutdown condition with the capability to place the unit in a safe shutdown condition. It is not the purpose of the SCR to allow continued operation of the plant at power. Egress from the MCR and access to the SCR is designed such that required minimum operations staffing can safely transition from the MCR to the SCR.

### **7.5.1 Main Control Room Use**

The MCR is designed with HSIs to support the following tasks:

1. Assessing the overall status and performance of the plant in any condition and providing necessary information to support operator actions
2. Monitoring and controlling fundamental safety functions
3. Monitoring the status and trends of key plant parameters (such as reactor power and rates of power change)
4. Operating the plant safely during all operational states, automatically or manually
5. Taking measures to maintain the plant in a safe state or to bring it back into a safe state after design basis events and design extension conditions
6. Maintaining the plant within the specified limits and conditions for the parameters associated with plant systems and equipment
7. Monitoring for failure of critical instrumentation and equipment
8. Confirming safety actions for the actuation of safety systems are automatically initiated when needed and that the relevant systems perform as intended
9. Determining the need and the time for manual initiation or intervention of specified safety actions
10. Implementing emergency operating procedures, emergency mitigating equipment guidelines, and severe accident management guidelines

The MCR is designed to be optimal for nominal shift complement of the control room operator role-holders. The MCR at concept stage is sized and designed to accommodate the expected maximum number of people on a continuous basis. This maximum number will be based on the MCR nominal staffing plus six other people. The basis for six additional people is to allow sufficient space for the supervisory role and five others (e.g., trainees, observers, etc.). The capability for six additional people also provides the necessary facilities to support plant conditions and evolutions that require more than the normal operations complement, for example during pre-operational and startup testing. The size and layout of the MCR are expected to be adjusted as the design progresses as a result of the staffing analysis being conducted as part of the HFE program of activities, as described in Chapter 18, Subsection 18.2.4. The required MCR personnel have workstations designed to support their specific control, information, communication, and work coordination needs. Each MCR workstation also contains space for a

role trainee. The MCR is designed in accordance with current international best practice codes and standards for control room design, integrating results from HFE analyses and specified HFE design requirements as described in Chapter 18, Section 18.3.

### **7.5.2 Main Control Room Layout**

Figure 7.5-1 represents the concept for the MCR. This concept is expected to evolve with the program of safety analyses and HFE activities being performed as an integral part of the BWRX-300 design. The analyses are used to develop an MCR design that optimally supports required personnel roles for all plant conditions.

Key to the MCR layout design is that tasks take place in a position that does not result in blocking views of the Group-View Display. The arrangement of the workstations in the MCR also facilitates communication between all MCR personnel, including direct conversation and visual contact. MCR design also fosters the oversight and command and control responsibilities of the required supervisory roles. The layout of workstations provides sufficient clearance for foot traffic and maintenance between consoles. The layout of workstations provides enough space for the proper storage, placement, and use of tools and procedures in the MCR. The workstations are generally designed for seated operations; however, it is expected that the HFE analysis activities described in Chapter 18 will identify the correct configuration and height required to ensure optimal visibility of individual and shared workspaces within the MCR.

The MCR Group-View Display is an array of large VDUs displaying a consolidated overview of information about plant systems. The purpose of the Group-View Display is to support cohesive situational awareness and crew coordination through shared monitoring of key parameters and alarm information relating to overall plant status, including emergency operating procedure entry conditions.

The Group-View Display VDUs are located and designed to support optimal line-of-sight and viewing angles by the necessary subset of MCR users. Group-View Display and their content are designed and sized to support monitoring and readability of on-screen component labels and indications to the required MCR users. Group-View Display content can be managed at the supervisory workstation via mouse and keyboard interface, including access to closed circuit televisions. Group-View Display VDUs do not have plant control functions, as these are designed for monitoring purposes only. Group-View Display alarms from SC1, SC2, SC3, and Non-Safety Class systems are also included on the Group-View Display VDUs. Additional information about alarms will be accessible at the workstation. The Group-View Display connected to the DCIS are also available at the workstation VDUs. The purpose of the Group-View Display is to be a supplemental, not a sole source, of information.

The MCR operator workstation is made up of three sections: two SC3 workstation sections and one SC1 workstation section. The MCR operator workstations contain the controls and indications needed to perform the assigned MCR tasks. VDU displays are designed using the HFE processes described in Chapter 18, Subsection 18.3.5, and provide easy and simple navigation to available parameters and controls.

In the MCR concept, the SC1 workstation contains the necessary divisionally dedicated VDUs for each of the three SC1 divisions, for a total of six VDUs. The VDUs are divisionally separated so that information from one division does not populate VDUs of another division. Some VDUs are used primarily for monitoring parameters and overall system status indications and for use as a backup if a VDU becomes nonfunctional. Other VDUs are used primarily for controls and confirmatory indications that the control has been actuated, a control signal has been sent, and the control action is having the intended impact on the system. SC1 VDUs are controlled by a

touchscreen interface. Hardware and software-based controls are positioned in compliance with specified HFE design requirements and to preserve sufficient laydown areas for procedures.

The SC3 workstations will contain the necessary VDUs to support monitoring, control, and alarm management of all other plant functions. These VDUs are controlled by an appropriate number of wired keyboards and mouse interfaces. SC3 workstation VDUs are capable of displaying any provided plant and system information, as supported by the I&C architecture. These VDUs only allow for control of SC3 and Non-Safety categorized functions and does not allow more than one operator to control the same equipment at the same time. Non-SC1 VDUs allow for flexible use and are not divisionally separated or dedicated.

MCR SC3 workstation VDUs are sized and positioned in compliance with specified HFE design requirements for viewing angles for frequently used indications and alarms, to support readability, and to preserve line-of-sight to Group-View Display. The arrangement of the SC3 workstations in the MCR facilitates verbal communication.

The MCR SC3 workstations also contain a VDU (each with a mouse and keyboard input device) dedicated for business Local Area Network (LAN) and one of the SC3 workstations contain a VDU (with a mouse and keyboard input device) for the fire protection system. The business LAN has no connection to and is physically separate from the plant I&C networks. The SC3 workstation also contains space for communications equipment, procedure laydown, and administrative tasks.

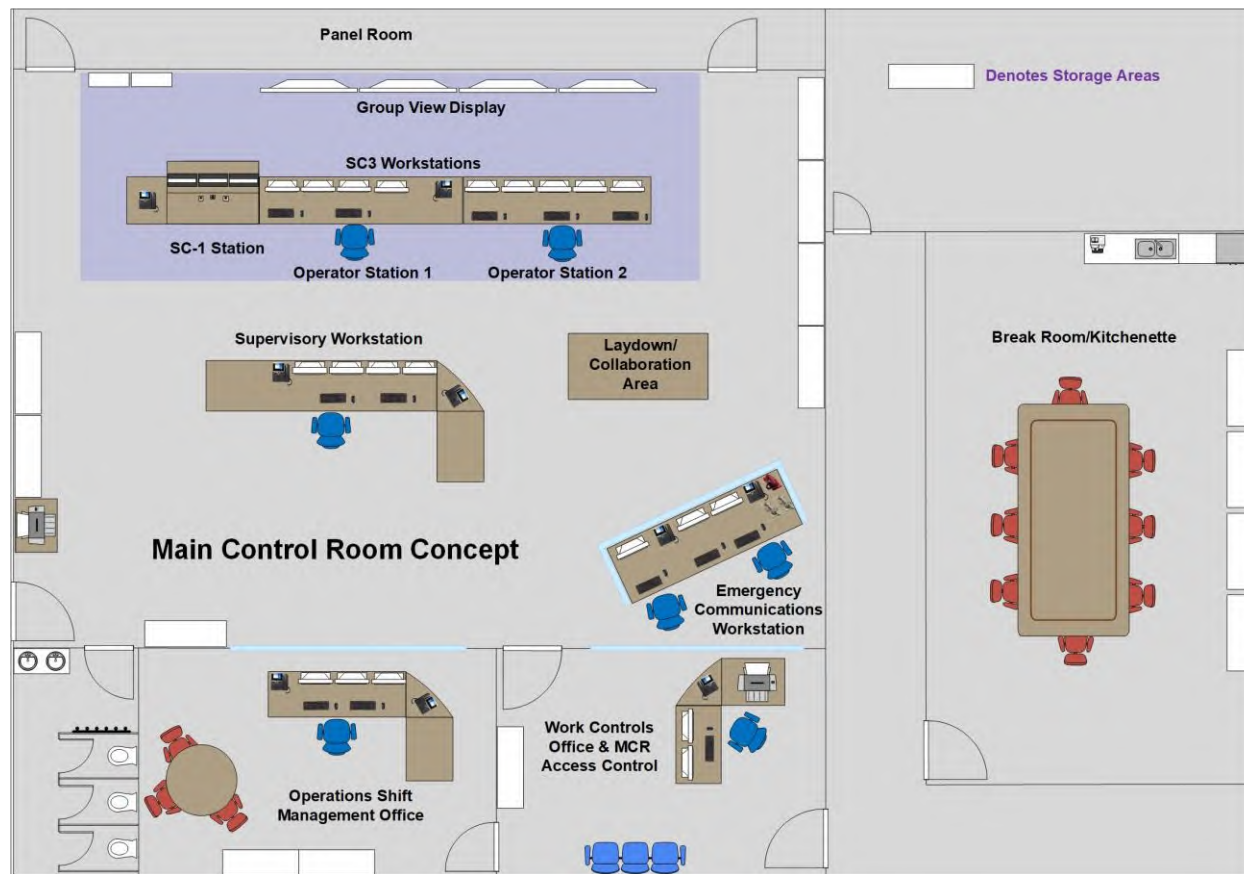
The MCR supervisory role workstation is implemented with SC3 equipment and designed with a work area is large enough to accommodate three seated people. The MCR supervisory workstation is located behind the SC3 workstation; it is suitably positioned to maintain a clear line-of-sight to the SC3 workstations and support verbal communication with the other workstations in the room. The supervisory workstation is also positioned to maintain a clear line-of-sight to the Group-View Display. The MCR supervisory workstation contains at least two VDUs with indications (no control) to monitor operations as they are being performed. The MCR supervisory workstation VDUs are controlled by a wired keyboard and mouse interface. The MCR supervisory workstation also contains two VDUs dedicated to the business LAN, as well as space for communications equipment, procedure and drawing laydown, and administrative tasks.

The MCR also contains an emergency communications workspace. The intent of the workspace is to allow designated personnel as required during events to communicate externally to the MCR, in such a way as to not disrupt the personnel focused on monitoring and controlling the plant in response to the event. The emergency communications workspace is located and designed with suitable provisions to minimize disturbance to the operating crew. The emergency communications workstation contains at least one VDU with indications (i.e., no control) from all Safety Class and Non-Safety Class systems for monitoring plant conditions. The emergency communications workstation also contains at least two VDUs to access the business LAN. The emergency communications workstation VDUs are controlled by a wired keyboard and mouse interface. VDUs are sized and positioned in compliance with specified HFE design requirements. The emergency communications workspace also contains space for all required communications equipment and administrative tasks.

### **7.5.3 References**

None.

NEDO-33956 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 7.5-1: MCR and Surrounding Areas (Top View)**

## **7.6 Instrumentation and Control in Secondary Control Room**

In the current concept design, the SCR is located in the Reactor Building. The routes from the MCR to the SCR are described in Chapter 6, Section 6.4. The SCR includes the required HSI that enable operators to perform the defined set of functions required for responding to the identified plant events and conditions for which the MCR cannot be used.

The SCR is designed in accordance with current international best practice codes and standards for control room design, integrating results from HFE analyses and specified HFE design requirements as described in Chapter 18, Section 18.3.

### **7.6.1 Control Transfer Function**

The necessary MCR to SCR control transfer functions are located in a suitable location determined through HFE analysis, either in the SCR or accessible via the qualified access path. Access to the SCR is indicated by alarms in the MCR to ensure unauthorized access is detected. The SCR includes all HSI inventory required to maintain the plant in a safe state for scenarios requiring MCR evacuation.

The necessary MCR to SCR control transfer provisions ensure the availability of necessary HSI after MCR evacuation and access control when the SCR is not activated. Design features that prevent or mitigate spurious actuations due to fire are described in Chapter 9, Subsection 9A.6.6.

### **7.6.2 Secondary Control Room Use**

The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable. The required functions are derived as a result of safety and HFE analyses.

The SCR includes suitable facilities for habitability and well as workspace for tasks to support required usage. The SCR contains a suitable supply of food and water. The SCR also contains adequate space and provisions for sleeping as required by the postulated scenarios in which it is used.

The SCR is designed to accommodate the expected staffing based on HFE staffing analysis for the expected usage conditions, as described in Chapter 18, Subsection 18.2.5. A suitable number of workstations are provided to support the specific task, communication, and work coordination needs for expected personnel.

### **7.6.3 Secondary Control Room Layout**

The layout of the workstations and HSI in the SCR provides the personnel with adequate information to assess the plant state and perform actions maintain the plant in a safe state, if required for the expected usage scenarios.

To reduce the likelihood of human errors and time needed to resume monitoring and control tasks within the SCR, the layout and HSI of the SCR are design to be consistent with the MCR to the extent possible, with differences driven by different purpose and user task needs.

The SCR operator workstation can accommodate two people. The SC1 workstation contains the required HSI to perform the functions expected to be performed in the SCR.

The SCR SC3 workstation section contains two VDUs to support monitoring, control, and alarm management. The SC3 workstation VDUs can display any provided plant and system information, as supported by the I&C architecture, but will only allow for control of SC3 and Non-Safety categorized functions. SC3 VDUs allow for flexible use and are not divisionally separated or dedicated. These VDUs are controlled by a wired keyboard and mouse.

SCR SC3 workstation VDUs are sized and positioned in compliance with specified HFE design requirements for viewing angles for frequently used HSI components. The SCR SC3 workstation suitably positioned to support effective verbal communication with the other workstations in the room. The SCR SC3 workstation also contains space for communications equipment, procedure laydown, and administrative tasks.

The SCR supervisory workstation is implemented with SC3 equipment and large enough to accommodate two people. The supervisory workstation is designed and positioned to maintain a clear line-of-sight to the SCR SC1 and SC3 workstations. The SCR supervisory workstation suitably positioned to maintain a clear line-of-sight to the SC1 and SC3 workstations and support effective verbal communication throughout the room. The supervisory workstation contains two non-safety classified VDUs to provide SPDS indications (i.e., no control) to monitor plant conditions, and a VDU or tablet is provided for business LAN connection. The business LAN has no connection to and is physically separate from the plant I&C networks. The SCR supervisory workstation also contains space for communications equipment, procedure and drawing laydown, and administrative tasks.

All VDUs are controlled by a wired keyboard and mouse interface except, where applicable or dictated by safety requirements. The SC1 VDUs will be a touchscreen type interface. If supplied, tablets are controlled by a touchscreen interface.

The SCR has an emergency communications workspace located and designed with suitable provisions to minimize disturbance to the operating crew. The emergency communications workstation contains VDUs or Wi-Fi-enabled tablets for connection to business LAN with at least one VDU with displaying SPDS indications (i.e., no control) to monitor and report externally on plant conditions. The emergency communications workspace also contains space for communications equipment, procedures, and drawings in support of performing administrative tasks.

#### **7.6.4 References**

None.

## **7.7 Instrumentation and Control in the Emergency Response Facilities**

The BWRX-300 MCR is the primary location for plant monitoring and control during normal, abnormal, and most emergency conditions. The I&C capabilities in the MCR are described in Section 7.5. The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable. The I&C capabilities in the SCR are described in Section 7.6.

The Emergency Preparedness and Response capabilities and associated facilities are described in Chapter 19. The Emergency Operations Centre (EOC) is located on-site near the control room. The Site Management Centre is located on-site, but outside of the protected area. It is provided for use by technical support staff and emergency support staff in the event of an emergency. The EOC and Site Management Centre contain the equipment necessary to display data needed in the facilities to analyze plant conditions. This facility is staffed with on-call teams of Emergency Response Organization personnel that can provide technical support.

An Emergency Response Facility (ERF) is located off-site and outside of the plume exposure planning zone. The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable.

The plant information necessary to support overall emergency response from the MCR, SCR, EOC, Site Management Centre, and ERF is provided either directly via the Unit Data Highway or the Plant Data Highway unidirectional boundary device interface shown in Figure 7.2-1, depending on cyber security requirements and I&C requirements required for to each location.

The MCR emergency communications workspace contains space for communications equipment and administrative tasks. The plant communication systems are described in Chapter 9A, Section 9A.9.1.

### **7.7.1 References**

None.

## **7.8 Hazard Analysis for Instrumentation and Control Systems**

Hazards for the BWRX-300 I&C systems are addressed in five ways. First, specific system requirements are developed based on plant safety analysis (i.e., plant-level hazards), as described in Chapter 15. These are the actuation functional requirements implemented by DL2, DL3, and DL4a. Specific safety functions actions are mode dependent, as specified by the Mode Switch position shown for each function in Tables 7.3-1, 7.3-3, and 7.3-4. Second, specific hardware and software technology hazards for the digital I&C equipment (i.e., digital technology hazards) as addressed by system level failure modes and effect analyses performed using the methods described in IEC 60812 (Reference 7.8-1). Third, a rigorous development process, as described in Section 7.4, is used to minimize human errors during system development and operation (i.e., systematic hazards). Fourth, the I&C equipment is designed to provide protection against external and internal hazards, as described in Chapter 3, Sections 3.3 and 3.4. And fifth, the I&C equipment is qualified, as described in Chapters 3, Section 3.9, and Section 7.3.

### **7.8.1 References**

- 7.8-1 IEC 60812, "Failure modes and effects analysis (FMEA and FMECA)," International Electrotechnical Commission.





**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33957

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 8  
Electrical Power**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AOO	Anticipated Operational Occurrence
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
DL	Defense Line
EDS	Electrical Distribution System
FMCRD	Fine Motion Control Rod Drive
GEH	GE Hitachi Nuclear Energy
GSU	Generator Step Up Transformer
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
LOOP	Loss-of-Offsite Power
LV	Low Voltage
MOD	Motor Operated Disconnect
MV	Medium Voltage
OPG	Ontario Power Generation
PPS	Preferred Power Supply
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RG	Regulatory Guide
RVT	Regulating Voltage Transformer
SBO	Station Blackout
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Category
SDG	Standby Diesel Generator
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

UAT	Unit Auxiliary Transformer
UPS	Uninterruptable Power Supply
VAC	Volts Alternating Current
VDC	Volts Direct Current

## TABLE OF CONTENTS

8.0	ELECTRICAL POWER SYSTEMS .....	8-1
8.1	Description of the Electrical Power System.....	8-1
8.1.1	Licensing Requirements .....	8-1
8.2	General Principles and Design Approach .....	8-5
8.2.1	General Principles .....	8-5
8.2.2	Electrical System Design Approach .....	8-5
8.3	Offsite Power Systems .....	8-7
8.3.1	Safety and Equipment Functions .....	8-7
8.3.2	Safety Design Bases .....	8-7
8.3.3	Description .....	8-7
8.3.4	Materials.....	8-8
8.3.5	Interfaces With Other Equipment or Systems .....	8-8
8.3.6	System and Equipment Operation .....	8-8
8.3.7	Instrumentation and Control.....	8-8
8.3.8	Monitoring, Inspection, Testing, and Maintenance .....	8-8
8.3.9	Radiological Aspects .....	8-9
8.3.10	Performance and Safety Evaluation.....	8-9
8.4	On-site Power Systems .....	8-10
8.4.1	On-site Alternating Current Power Systems.....	8-10
8.4.2	On-site Direct Current Power Systems .....	8-17
8.5	Electrical Cables and Raceways.....	8-24
8.5.1	System and Equipment Functions .....	8-24
8.5.2	Safety Design Bases .....	8-24
8.5.3	Description .....	8-25
8.5.4	Materials.....	8-25
8.5.5	Interfaces With Other Equipment or Systems .....	8-25
8.5.6	Instrumentation and Control.....	8-25
8.5.7	Monitoring, Inspection, Testing, and Maintenance .....	8-25
8.5.1	Radiological Aspects .....	8-25
8.5.2	Performance and Safety Evaluation.....	8-25
8.6	Grounding, Lightning Protection and Electromagnetic Compatibility .....	8-26
8.6.1	System and Equipment Functions .....	8-26
8.6.2	Safety Design Basis.....	8-26

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

8.6.3	Description .....	8-26
8.6.4	Materials .....	8-28
8.6.5	Interfaces With Other Equipment or Systems .....	8-28
8.6.6	System and Equipment Operation .....	8-28
8.6.7	Instrumentation and Control.....	8-28
8.6.8	Monitoring, Inspection, Testing, and Maintenance .....	8-28
8.6.9	Radiological Aspects .....	8-29
8.6.10	Performance and Safety Evaluation.....	8-29
8.7	Main Equipment Types.....	8-30
8.7.1	System and Equipment Functions .....	8-30
8.8	References .....	8-31

**LIST OF TABLES**

None.



**LIST OF FIGURES**

Figure 8.8-1: Main Electrical One-Line .....	8-33
Figure 8.8-2: Safety Class 1 Low Voltage Uninterruptible Power Supply Systems .....	8-34
Figure 8.8-3: Safety Class 3 Low Voltage Busses A21 and B21 .....	8-35
Figure 8.8-4: Safety Class 2 Low Voltage Uninterruptible Power Supply Systems .....	8-36
Figure 8.8-5: Safety Class 2 Low Voltage Uninterruptible Power Supply Distribution .....	8-37
Figure 8.8-6: Non-Safety Class Medium Voltage Busses A1-A2 and B1-B2.....	8-38
Figure 8.8-7: Typical BWRX-300 Switchyard Interface.....	8-39
Figure 8.8-8: Non-Safety Class 125 VDC System .....	8-40
Figure 8.8-9: Non-Safety Class 208/120 VAC System .....	8-41

## **8.0 ELECTRICAL POWER SYSTEMS**

This chapter describes the BWRX-300 plant electrical system and provides a description of the BWRX-300 electrical system requirements. Design Requirements, and details of the electrical systems as described in this chapter are derived from and are consistent with plant requirements, design decisions and assumptions documented in references listed in Section 8.8 below. The BWRX-300 Electrical Distribution System (EDS) design complies with Canadian national requirements as described in this document. In addition, provincial requirements within Canada vary and require evaluation for each provincial area.

The EDS architecture is a configuration of generators, busses, transformers, and load centers that supply power to the design loads. The design described in this document is for a 60 Hz Alternating Current (AC) power system, with 4.16 kV for the MV level and 600 Volts Alternating Current (VAC) for the Low Voltage (LV) level.

In addition to describing the electrical systems and associated requirements, this chapter includes plant one-lines that depict the switchyard, switchyard interface, and the MV and LV AC distribution systems, which includes the DC battery systems.

### **8.1 Description of the Electrical Power System**

The BWRX-300 EDS is an integrated power supply and distribution system for the power plant. Because of the integration, this architecture and requirements document describes the three plant systems that constitute the overall electrical system. The three plant systems are grouped based on safety classification.

The three plant systems have appropriate levels of hardware and software quality corresponding to the functions and systems receiving power. The systems provide reliable power to various plant functions (electrical loads) and provide an electrical pathway for the main generator to the utility switchyard/grid. The electrical systems are appropriately protected, monitored, recorded, and controlled by the plant operators. Various segments of the plant electrical system can operate independently.

Figure 8.8-1 depicts the main BWRX-300 plant electrical one-line that implements the above functions and classification and meets the requirements identified in Subsection 8.4.1 (note the Safety Class (SC) boundaries shown on Figure 8.8-1). The Safety Class 1 (SC1) EDS, Safety Class 2 (SC2) and 3 EDS, and Non-Safety Category (SCN) EDS are described in greater detail in the following sections.

#### **8.1.1 Licensing Requirements**

The BWRX-300 electrical system design meets the requirements of CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 8.8-8), Sections 7.10 and 8.9, and complies with the requirements except as indicated. The BWRX-300 electrical system is also designed to meet International Atomic Energy Agency (IAEA) requirements.

Although the BWRX-300 shutdown and heat removal functions can be initiated by active SC1 and SC2 and 3 control systems, loss-of-offsite and on-site electrical power, including battery-backed Uninterruptable Power Supplies (UPSs), automatically causes control rod insertion/plant shutdown and initiates decay heat removal. In any Design Basis Accident (DBA) and Beyond Design Basis Accident, electricity is required for monitoring, alarms, and communications. The heat removal functions can be provided actively by the Standby Diesel Generators (SDGs), which can operate autonomously for one week before requiring refueling.

Per CNSC REGDOC-1.1.2 (Reference 8.8-9), Section 4.5.10, the requirements specified in CNSC REGDOC-2.5.2 (Reference 8.8-8), Sections 7.10 and 8.9 are to be addressed. These requirements for the BWRX-300 electrical systems are discussed below.

#### **8.1.1.1 REGDOC-2.5.2 Section 7.10 Requirements**

CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 7.10 defines requirements for safety support systems.

The BWRX-300 electrical system meets the requirements of CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 7.10 with normal plant power provided via the main generator or offsite power, backup power provided via SDGs, and SC1 emergency power provided by UPSs. The electrical system is designed with sufficient capacity to meet load requirements to support Fundamental Safety Functions. The design incorporates an appropriate Defence-in-Depth strategy to ensure availability and reliability of the supported systems.

For BWRX-300, the SC1 EDS provides at least 72 hours of battery-backed AC and DC power, after which it is supported by the SDGs for at least one week, or alternatively, is supported by connection of portable generators. In a complete loss-of-offsite/generator AC power, batteries are charged via the SDGs or the portable generator connections while also powering the required loads for safety functions.

Capacity margin for the batteries is established by meeting the IEEE Std 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications" (Reference 8.8-20) battery age, temperature, and load margin requirements.

The emergency support systems have been designed to:

- Be independent of normal and backup system
- Support continuity of the Fundamental Safety Functions until long-term (normal or backup) service is re-established:
  - Without the need for operator action to connect temporary on-site services for at least eight hours
  - Without the need for offsite services and support for at least 72 hours
- Have a capacity margin that allows for future increases in demand
- Be testable under design load conditions, where practicable

#### **8.1.1.2 REGDOC-2.5.2 Section 8.9 Requirements**

CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 8.9 defines requirements for electrical power systems.

As described in Subsection 8.4.2.2, the standby and emergency power systems are represented in the SC1 EDS and the SC2 and 3 EDS. The SC2 and 3 EDS encompasses the Defense Line 4A (DL4a) (SC2) and DL2 (Safety Class 3 (SC3)) components, and the SC1 EDS encompasses the DL3 (SC1) components. The design of these systems ensures sufficient capacity to support associated safety functions, and that availability and reliability of the system is commensurate with the safety significance of the connected loads.

The BWRX-300 electrical systems are designed to accommodate grid disturbances within the range of those shown to be possible by transmission operator studies at the point of interconnect. Such studies include maximum and minimum transient and steady-state grid voltages, available fault current, and consideration of the effect of geomagnetic induced currents resulting from coronal mass ejections. The design includes appropriate mitigations based on the results of

studies such as the application of automatic load tap changes, surge suppression, transformer zero sequence current monitoring, and blocking capacitors.

However, in contrast to the guidance of CNSC REGDOC-2.5.2 (Reference 8.8-8), the BWRX-300 design is not required to include two or more independent connections between the plant and the grid because the BWRX-300 plant does not require offsite electrical power to achieve safe shutdown, nor does it require any type of electrical power to protect the public from Severe Accidents. All SC1 power is provided by three divisions of SC1 batteries and UPSs. The batteries are sized for 72 hours, which allows for either of the two SC3 SDGs to start, external portable generators to be connected, or offsite power to be restored. Although the BWRX-300 is not committed to the requirement in the aforementioned regulatory documents that the plant have two offsite power sources, two sources of offsite power are typically used to improve reliability and operational flexibility, as shown in Figure 8.8-1 and Figure 8.8-9.

#### **8.1.1.3 REGDOC-2.5.2 Section 8.9.1 Requirements**

CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 8.9.1 further defines design requirements for electrical power systems with a focus on standby and emergency power systems.

The BWRX-300 electrical systems are designed to meet CNSC REGDOC-2.5.2 (Reference 8.8-8) requirements for standby and emergency power systems. Although the BWRX-300 does not require any on-site or offsite AC power for safety, the permanently installed SDGs provide standby AC power and generally meet all requirements of the referenced section of the REGDOC. Restoring power as soon as possible is desirable. Following a plant Loss-of-Offsite Power (LOOP), the SDGs are expected (although not credited) to automatically start and achieve rated voltage and frequency within 30 seconds. The completion of automatic sequencing is expected (but not credited) to take no longer than two minutes. The permanently installed batteries provide the emergency power required for the SC1, SC2, and SC3 loads to meet plant needs for 72-hours. If offsite power is not restored, AC power is provided via the SDGs or via permanently installed external connections for use with portable generators.

#### **8.1.1.4 REGDOC-2.5.2 Section 8.9.2 Requirements**

CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 8.9.2 further defines requirements for electrical power systems with a focus on DC and uninterruptible power systems.

The BWRX-300 electrical systems are designed to meet CNSC REGDOC-2.5.2 (Reference 8.8-8) requirements for DC and uninterruptible power systems. The BWRX-300 DC systems use two battery chargers, but only a single battery per division. The chargers are arranged such that one can carry the DC loads directly while the other applies an equalizing charge such that the loads do not have to be exposed to the equalizing voltage. It is important to note that this operational feature is optional and is not required to be used. Use of this feature removes the battery from the division. Additionally, the various DC loads are designed for equalizing voltages, if required.

Additional guidance in CNSC REGDOC-2.5.2 (Reference 8.8-8) recommends the uninterruptible power system include a normal AC supply to the inverter and a bypass AC supply with an automatic switchover. The BWRX-300 electrical system design takes exception to this guidance in the design of the SC1 EDS UPS system. While the divisional electrical systems use two redundant UPSs, the bypass line breakers are kept open, and automatic swap over to the bypass AC is not provided. The reason for operating without the bypass breaker closed is in response to the Forsmark event (U.S. NRC ML102070360, "Lessons from Forsmark Electrical Event" (Reference 8.8-10)).

**8.1.1.5 REGDOC-2.5.2 Section 8.9.3 Requirements**

CNSC REGDOC-2.5.2 (Reference 8.8-8), Section 8.9.3 defines requirements for electrical power systems with a focus on alternate AC power supplies.

The BWRX-300 electrical systems are designed to meet CNSC REGDOC-2.5.2 (Reference 8.8-8) requirements for alternate AC power supplies. Permanently installed electrical connections/disconnects are provided for connection of portable generators as required. On-site portable, transportable, or fixed power sources, or offsite portable or transportable power sources, or a combination of these, are provide for this purpose by the utility operator of the power plant.

## **8.2 General Principles and Design Approach**

### **8.2.1 General Principles**

The BWRX-300 is designed using conventional engineering design norms and analyses found in most all similar GEH BWR plants. Analysis techniques utilized are GEH proprietary but utilize standard conventional engineering design techniques. Thorough design analysis is performed to support the design document using approved methods. Design configurations are evaluated to ensure that compatibility with other design features are consistent, and do not create adverse conditions.

### **8.2.2 Electrical System Design Approach**

The BWRX-300 is designed using proven techniques for producing a safe and reliable nuclear power plant. The design approach is focused on safety, reliability, and quality. Specifically, the electrical system design approach considers the following:

1. The electrical system is designed to continue to provide its design basis function commensurate with its SC for the following primary initiating events:
  - a. Grid abnormalities including Grid Transients
  - b. (LOOP)
  - c. Station Blackout (SBO)
  - d. Anticipated Electrical Faults
  - e. Single Failure of SC electrical equipment

2. The impact of such events on the electrical system are described below:

- a. Grid Abnormalities Including Grid Transients

The Unit Auxiliary Transformer (UAT) and Reserve Auxiliary Transformer (RAT) contain automatic load tap changers. The load tap changers monitor the secondary voltage and adjust the transformer tap to maintain the distribution voltage at acceptable levels. For grid events that cannot be addressed by the load tap changers or other means, protective relaying isolates the preferred power source and the SC3 SDGs start and load on busses A21 and B21 as depicted in Figures 8.8-1 and 8.8-3.

- b. LOOP

For a LOOP the protective relaying isolates the preferred power source and the SC3 SDG start and load on busses A21 and B21. Redundant SDGs supply redundant load groups. SC3 equipment is directly supplied by the SDGs and associated busses. SC1 and SC2 equipment is indirectly supplied by the SDGs via battery chargers and UPSs. Each SDG is sufficiently rated to supply 100% of required SC loads for a minimum of seven days following a LOOP.

- c. SBO

During a complete loss of AC power including both SDGs, SC1 and SC2 equipment is served by SC1 and SC2 station batteries. Batteries are sized to provide 100% of the load of their associated SC for a period of 72 hrs. During an SBO event the SC1 power system is sufficient to support all DL3 safety functions for a period of 72 hrs.

d. Anticipated Electrical Faults

Anticipated electrical faults are selectively isolated by protective devices. Such isolation is accomplished by ensuring acceptable coordination between protective devices in accordance with the requirements of IEEE Std 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations" (Reference 8.8-13), IEEE Std 242, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)" (Reference 8.8-26), and IEEE Std 741, "IEEE Standard for Criteria for the protection of Class 1E Power Systems and Equipment in the Nuclear Power Generating Stations" (Reference 8.8-27).

Bus arrangement, load allocation, and equipment and cable separation ensures that for any single fault performance of the SC function will not be compromised.

e. Single Failure SC Electrical Busses and Components

SC busses and electrical equipment are designed to single failure criteria. The loss of any single bus, train or division will not compromise the ability to perform the SC function.

The electrical power system is designed for reliability. Key reliability design requirements include:

- The UAT and RAT are redundant and sized to carry 100% of the station load
- Medium Voltage (MV) busses and load allocation allow for the loss of any single bus without loss of power to SC busses while minimizing the impact to plant generation
- SC busses and equipment are redundant and physically separated

The SC1 power system is designed and procured such that a Common Cause Failure resulting in the loss of power to SC1 loads performing an active safety function is not credible. This is accomplished by:

1. SC1 power does not require offsite normal, or standby power sources to perform its function
2. The three divisions of SC1 power are physically and electrically separated in accordance with the requirements of IEC 60709, "Nuclear power plants - Instrumentation, control, and electrical power systems important to safety – Separation" (Reference 8.8-15) and IEEE Std 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits" (Reference 8.8-29)
3. SC1 power system equipment and software is sufficiently simple, of limited functionality, analyzed per the requirements of IEC 62671:2013 Section 7.3, and procured tested commensurate with the requirements of its SC

The portions of the electrical system credited with a safety function are organized by Defense Line and SC. A description of the different electrical system SCs including voltages and divisions is provided in other sections of this chapter.

SC systems design is demonstrated to be adequate to perform its function by ensuring compliance with design criteria through design review, analysis and final system testing and commissioning.

The electrical system is designed with spare capacity to support modifications and upgrades typically expected during the life of the station.

### **8.3 Offsite Power Systems**

Electric power from the utility grid to the plant on-site power system is provided by two preferred power circuits. The transmission system serves the main offsite power circuit (Normal Preferred Power), and the reserve offsite power circuit (alternate preferred power).

The BWRX-300 switchyard design is customer dependent, but the plant design allows significant flexibility. Because of the passive safety design of the BWRX-300 plant, only a single preferred power feed between the plant and switchyard is required. Therefore, the utility is only required to provide one transmission line to the BWRX-300 switchyard from offsite sources. However, two sources of offsite power are typically used to improve reliability and operational flexibility, as shown in Figure 8.8-7.

For Canadian applications, connection of the plant to the grid with only a single transmission line is consistent with the guidance of CNSC REGDOC-2.5.2 (Reference 8.8-8), for passive plants.

#### **8.3.1 Safety and Equipment Functions**

The offsite preferred power system is designed to provide a continuous source of power to the on-site power system throughout plant startup, normal operation (including shutdown), and abnormal operations. The offsite power system provides no credited safety function. As a result, the total LOOP results in no impact on nuclear safety.

#### **8.3.2 Safety Design Bases**

The BWRX-300 safety design does not require offsite power to be present to mitigate any DBAs and therefore has no safety design basis.

#### **8.3.3 Description**

A local switchyard will be built to consolidate power output from the BWRX-300 Small Modular Reactor (SMR) Facility and connect it with Ontario electrical power grid.

The local switchyard will be operated via the Main Control Room (MCR) in the SMR Facility. The SMR Facility has two high-voltage connections with the local switchyard at a 230kV voltage level. One line to output power from the Generator Step Up Transformer (GSU) and one line to supply the RAT. The local switchyard has two redundant 230kV connections with the transmitter.

The local switchyard is an indoor Gas Insulated Switchgear type, following a breaker and half arrangement with two redundant busses. The local switchyard is designed to have local and remote-control capability. The plan for the local Darlington New Nuclear Project switchyard is that is located North of the SMR Facility, East of the Extended Holt Rd and South of the CN Rail tracks. The local switchyard control and protection designs are coordinated with the SMR Facility controls and protections to meet IESO, NPCC and NERC codes and standards.

##### **8.3.3.1 Design Requirements for the Offsite Power Systems**

The offsite power system is designed to provide a continuous and reliable source of power to the on-site power system throughout plant startup, normal operation (including shutdown), and abnormal operations.

The switchyard, to which the main offsite circuits are connected, has two full-capacity main busses.

The switchyard main busses are designed such that any circuit breaker can be isolated for maintenance without interrupting service to any circuit.

Circuit breakers used in the switchyard and connected to the BWRX-300 plant are sized and designed in accordance with IEEE Std C37.06, "IEEE Standard for AC High-Voltage Circuit



Breakers Rated on a Symmetrical Current Basis – Preferred Ratings and Related Required Capabilities for Voltages Above 1000V” (Reference 8.8-28).

Disconnecting switches used in the switchyard and connected to the BWRX-300 plant are sized and designed in accordance with IEEE Std C37.30.1, “IEEE Standard Requirements for AC High-Voltage Air Switches Rated Above 1000 V” (Reference 8.8-11).

#### **8.3.4 Materials**

Materials are expected to be suitable and compatible for use in the switchyard environment. Appropriate Canadian standards (including any Canadian provincial standards) and appropriate international standards are met. The switchyard is a typical switchyard design with commercially available equipment designed for this type of service.

#### **8.3.5 Interfaces With Other Equipment or Systems**

Interfaces with the switchyard are hard I/O with no network connections.

The offsite power system connects to the plant through two transmission lines, one connected to the GSU and one connected to the RAT. Switchyard monitoring parameters available in the MCR include breaker position, voltages, currents, power, and alarms.

The AC and DC power required by the switchyard equipment are separated from plant on-site power systems.

#### **8.3.6 System and Equipment Operation**

Switchyard breakers for the plant connections include design features that allow for remote operation by the plant operators.

#### **8.3.7 Instrumentation and Control**

Switchyard Instrumentation and Control (I&C) is provided in the MCR to monitor and control the breakers that interface with the plant.

#### **8.3.8 Monitoring, Inspection, Testing, and Maintenance**

Monitoring of the offsite power system (i.e., switchyard parameters) (and the switchyard breakers) is made available to the plant operator and the grid operator through separate communication systems. Switchyard monitoring parameters available in the MCR include breaker position, voltages, currents, power, and associated alarms.

Most inspections are primarily visual or infrared but individual breakers can be isolated for more intrusive inspections if needed.

Required electrical system testing is governed by industry standards. The electrical systems are designed to allow required tests to be performed with minimal disturbance to the balance of the system. Specific steps and alignments necessary for the successful performance of testing are to be controlled by plant procedures.

The switchyard system has been designed to be upgraded if necessary, using conventional techniques for upgrading software, or upgrading equipment. Replacement of major equipment (transformers, Breakers, etc.) could require a significant outage to repair/modify/replace the equipment. The switchyard main busses are designed such that any circuit breaker can be isolated for maintenance without interrupting service to any circuit.

#### **8.3.9 Radiological Aspects**

The offsite electrical systems are not normally exposed to radiological environments. Accident doses are expected to be within the normal range of operating envelope (refer to Chapter 3, Section 3.9 for environmental conditions).

#### **8.3.10 Performance and Safety Evaluation**

While there are no safety functions for the offsite power system function, the offsite power systems provide net 300 MW of power to the grid and utilizes offsite power to power plant loads during normal and abnormal operating conditions. Therefore, no performance or safety evaluation is unnecessary. As stated previously, the total LOOP results in no impact on nuclear safety.

## **8.4 On-site Power Systems**

### **8.4.1 On-site Alternating Current Power Systems**

#### **8.4.1.1 System and Equipment Functions**

The on-site AC power system starts at the GSU and RAT Motor Operated Disconnects (MODs) and transitions from there to include the GSU, the main generator, the main generator circuit breaker, the UAT, the RAT and the rest of the plant AC distribution as depicted on Figure 8.8-1.

The on-site power system includes four 4.16 kV MV busses A1, A2 and B1, B2. The normal Preferred Power Supply (PPS) is connected through the UAT to busses A2 and B2. The alternate PPS is connected through the RAT to busses A1 and B1 and serves as backup to the UAT.

The A1 and B1 MV busses are normally supplied by A2 and B2 busses during power operation and provide SCN power required for unit operation.

The A2 and B2 MV busses supply power to LV 600VAC SC3 load centers A21 and B21. The A21 and B21 LV busses can be aligned to the alternate PPS via MV busses A1 and B1.

Both A21 and B21 have dedicated and physically separated SDGs.

The MV AC power level is 4.16 kV and the LV AC power level includes 600VAC, 240/120V and 208/120V.

In addition to providing normal AC power to SC1 electrical equipment, the SC3 AC electrical power system is designed to power both SC2 and SC3 loads. These loads and additional SCN plant investment protection loads are fed from the 600 VAC SC3 A21 and B21 LV busses. The A21 and B21 LV busses are configured as illustrated on Figure 8.8-3. The A21 and B21 LV busses are in fire-barrier separated rooms.

The A21 and B21 LV busses can be supplied by one of two upstream MV busses as shown in Figure 8.8-1. The normal preferred power source is via the UAT and GSU. The alternate preferred power source is via RAT by direct connection to the switchyard. The A21 and B21 LV busses are normally aligned through the MV busses to separate windings of the UAT. The A21 LV bus is normally supplied by MV Bus A2 connected directly to the UAT. The alternate supply to A21 via bus A1 is normally open via the associated A21 main breaker. The B21 LV bus is normally aligned in a similar configuration. In this configuration, faults on MV busses A1 and B1 do not remove power from the MV busses A2 and B2, comprising the normal supply to A21 and B21.

In addition to the preferred power sources from the SCN electrical power system, the A21 and B21 LV busses can also be supplied by SC3 SDGs. The SC3 SDGs are electric start and radiator cooled and do not require plant mechanical support services for operation. Either of the SDGs has sufficient capacity to support all SC1 loads (three divisions) and SC2 and SC3 loads (i.e., either SDG can support active decay heat removal). There is on-site fuel storage for each SDG for at least a week of full power at rated load. Refer to Chapter 9, Subsection 9A.7.3 (Description - Fuel Storage and Transfer System) and Chapter 3, Section 3.8 (General Design Aspects for Electrical Systems and Components).

The A21 and B21 LV SDG backed busses are in separate fire-barrier separated rooms. The A21 and B21 LV SDG backed busses and associated SDGs are located as close as reasonably possible to each other.

A21 and B21 busses provide normal power to the following SC1 and SC2 critical loads:

- SC1 UPSs and battery chargers
- the SC2 LGs
- SC2 supply Fine Motion Control Rod Drive (FMCRD) UPSs

A21 and B21 and derivative busses, SDGs, digital protection relays, and other components are designed, selected, and qualified to the applicable requirements of SC3 or SC2 as applicable

The loads assigned to the SC2 UPSs and battery chargers are required for the BWRX-300 to achieve cold shutdown, and include the power equipment that provides a diverse, active alternative to SC1 vessel injection and vessel and containment heat removal functions.

During a LOOP, the SC3 busses are automatically powered by the associated SDG. The SDG supply breakers to the SC3 busses are interlocked with the normal offsite source breakers to prevent the undesired paralleling of sources, while allowing periodic SDG testing by synchronizing to and loading into either normal power feed.

#### **8.4.1.2 Safety Design Basis**

The BWRX-300 has two connections to the switchyard. The normal preferred power source is connected to the primary side MODs of the GSU; the secondary side of the GSU is connected to the main generator breaker and includes a branch to the UAT of the plant. The alternate preferred power source is connected to the MODs at the RAT primary side. The UAT and RAT secondaries are connected to the plant MV busses. The on-site power systems are not designed to perform any safety functions to mitigate a DBA. The batteries are designed to provide power to post accident monitoring equipment to the control rooms for at least 72 hours.

#### **8.4.1.3 Description**

As discussed previously, the BWRX-300 has two connections to the switchyard. The normal preferred power source is connected to the primary side MODs of the GSU; the secondary side of the GSU is connected to the main generator breaker and includes a branch to the UAT of the plant. The alternate preferred power source is connected to the MODs at the RAT primary side. The UAT and RAT secondaries are connected to the plant MV busses. The configuration is shown in Figure 8.8-1.

The GSU provides transmission of the main generator output to the grid when the plant is online and can provide power to the plant when the main generator is offline via backfeed. When the generator breaker is closed and the plant is synchronized with the grid, the generator gross power output is approximately 300 MWe. When the generator breaker is opened, the switchyard can “backfeed” power to the plant through the GSU and the UAT. The UAT is sized to support 100% of the plant normally operating power generation SCN loads and SC (SC1, SC2, and SC3) loads. The UAT is protected by breakers on both the primary and secondary side. Protection is provided for fault conditions and includes both transient and steady-state overvoltage protection. If a UAT fault occurs, the primary side breaker opens to prevent drawing fault current from the main generator and the grid.

The RAT interface provides an alternate power path from the grid to the plant power distribution system. Like the UAT, the RAT is sized to support 100% of the plant normally operating power generation and SC loads. The purpose of the alternate preferred power source is to provide operational flexibility and reliability to the system. For example, in the event of a failed UAT during power operation, the system isolates the failed UAT and automatically fast transfer to the RAT while the plant continues to operate providing power to the switchyard.

Although the plant transformers are to be monitored by a condition monitoring system, it is expected that the transformer life is approximately 30 years. The design specifically supports changing the transformer during a long outage in 25 – 30 years before the transformer end of life.

The UAT and RAT are equipped with automatic load tap changers that assure the plant busses maintain nominal voltage under all grid or main generator voltage conditions and when the house loads change from power operation to outages. All other on-site transformers in load centers have manual tap changers. However, after initial setup, these should not have to be readjusted.

Figure 8.8-6 depicts the A1 and B1 MV bus configuration for the SCN EDS. The bus configuration and load allocation are such that these busses can be shut down for maintenance during outages and can be shut down one at a time in normal operation while minimizing the impact to plant power generation.

With the plant online providing power to the switchyard, the MV busses A1, A2, B1, and B2 busses are normally aligned to the UAT. With the plant offline, the MV busses are normally aligned to the RAT. The MV busses could also be aligned to the RAT during power generation, either manually or via fast transfer. Another possible alignment is with the plant offline supplying power to the MV busses via the UAT and GSU with the generator breaker open in a backfeed configuration.

The MV busses A1, A2, B1, and B2 protective relays are designed to automatically fast bus transfer between the UAT and the RAT. The fast bus transfer is automatically initiated on loss of power to busses A2 and B2, a faulted circuit in the normal PPS, and on a unit trip. The intent of this automatic action is to minimize the potential for a loss of power to the plant EDS and to minimize entrance into the backfeed operation alignment. The transfer logic is configured in such a way that it only occurs if power is available from the RAT. This prevents a transfer to a dead source. Additionally, to avoid the possibility of cycling between the normal and alternate preferred source, the fast bus transfer is one-directional (i.e., there is no possibility of an automatic transfer from the alternate preferred to the normal preferred source).

#### ***8.4.1.3.1 System Design Requirements for On-site AC Power***

The generic system design requirements for the AC electrical systems are listed below:

1. The LV AC distribution systems are solidly grounded.
2. Load assignments within the plant electrical systems are evenly distributed between the MV busses, to the extent practical, to maintain a reasonable load balance on the UAT and RAT secondary windings.
3. Current-carrying cables in the plant electrical systems are qualified to meet the IEEE Std 383, "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities" (Reference 8.8-21) requirements for cables and splices for nuclear facilities.
4. The EDS equipment can perform required safety functions when exposed to worst-case environmental conditions of the applicable room/building.
5. The equipment and components in the plant electrical systems are classified to the seismic category appropriate for the safety classification and location of the system.
6. The equipment and components in the plant electrical systems are mounted to maintain the applicable seismic qualification.
7. Standby lighting with a Safety Category 1, 2, or 3 function are powered from busses backed by a standby power source of the same (or higher) safety classification.
8. Normal lighting is powered from the SCN power generation busses to the extent practical.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

In addition to the generic system design requirements listed above, the following design requirements are applicable for the SCN EDS:

1. The UAT is sized to support 100% of the station loads for the worst-case operating scenario.
2. The UAT has a dedicated circuit breaker connected to the primary winding.
3. The UAT includes two secondary windings, both rated at 4.16 kV.
4. The UAT winding configuration is a Delta primary, with two Wye connected secondaries, connected such that high voltage (HV) winding Leads the LV winding by 30 degrees, as described in in IEEE C57.12.00 -2021.
5. The UAT Wye windings neutral bushings are low resistance grounded, as defined by IEEE 142-2007.
6. The UAT includes an automatic load tap changer, with a sensing voltage corresponding to the average voltage of the secondary feeds.
7. The main generator circuit breaker is sized and designed in accordance with IEEE/IEC 62271-37-013-2021.
8. The GSU is comprised of three in-service, single-phase transformers, with one installed single-phase, spare transformer.
9. The installed spare transformer for the GSU is maintained ready for operation with support systems powered such that it can be placed into service within a few days.
10. The GSU winding configuration is a generator connected Delta, with a switchyard connected Wye, connected such that HV winding lagging the LV winding by 30 degrees, as described in IEEE C57.12.00-2021.
11. The GSU Wye windings neutral bushings are solidly grounded.
12. The RAT is sized to support 100% of the station loads for the worst-case operating scenario.
13. The RAT has a dedicated MOD connected to the primary winding.
14. The RAT includes two secondary windings, both rated at 4.16 kV.
15. The RAT winding configuration is a Wye primary, with two Wye connected secondaries.
16. The RAT primary winding neutral bushing is solidly grounded.
17. The RAT secondary Wye windings neutral bushings are low resistance grounded as defined by IEEE 142-2007.
18. The RAT includes an automatic load tap changer, with a sensing voltage corresponding to the average voltage of the secondary feeds.
19. SCN EDS MV busses are used to supply SCN loads required exclusively for power generation.
20. Fast bus transfer between the normal and alternate preferred power source is initiated when bus voltage is lost to the A2 and B2 busses, during a faulted normal preferred power circuit, and during a unit trip.
21. The fast bus transfer is sufficiently fast to prevent downstream loads from dropping off the bus.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

22. The fast bus transfer logic prevents transfer to a dead bus.
23. The fast bus transfer is one-directional from the normal preferred source to the alternate preferred source.
24. Unit substation transformers winding configuration is Delta primary to Wye secondary.
25. Unit substation transformers secondary Wye winding is solidly grounded.

In addition to the generic system design requirements listed above, the following design requirements are applicable for the SC2 and 3 EDS:

1. Provides redundant LV power feeds to each SC1 division, with each feed designed to provide the full power of the respective division
2. Two SDGs provide backup power to the LV A21 and B21 busses for plant mechanical systems, investment protection, and the plant DCIS
3. Each SDG is sized to supply 100% of the required SC1 divisional loads in all three divisions while also supplying power to the required SC2 and SC3 loads during a LOOP event
4. The SDGs provide adequate voltage and frequency regulation during load sequencing and steady-state conditions
5. The SDGs can accept a step change corresponding to the size of the largest load and remain within frequency and voltage limits
6. The SDGs are controllable from the MCR, the Secondary Control Room and locally
7. Once started, SDGs are capable of synchronizing to the plant electrical system for test purposes
8. The SDGs operate without requiring any plant services other than fuel, with the capability of continuous operation for at least one week without any offsite resources
9. The SDGs incorporate protective relaying for protection and control

The on-site AC power systems have appropriate levels of hardware and software quality corresponding to the functions and systems receiving power. The systems provide reliable power to various plant functions (electrical loads) and provide an electrical pathway for the main generator to the utility switchyard/grid. The electrical systems are appropriately protected, monitored, recorded, and controlled by the plant operators. Various segments of the plant electrical system can operate independently.

#### **8.4.1.4 Materials**

Materials to be used are specified and evaluated by GEH for design and commissioning. OPG is responsible for any changes following commissioning and operation. Materials are expected to be suitable and compatible for use in the environment. Appropriate Canadian standards (including any Canadian provincial standards) and appropriate international standards are met. The design utilizes materials typical for this type of service with commercially available equipment.

Materials are associated with the SC of the individual systems. Structures, Systems, and Components (SSC) are evaluated based on the SC and functions. Materials are evaluated and selected based on the environmental conditions and SC of the equipment as discussed in environmental conditions and SC of the equipment as discussed in Chapter 18.

#### **8.4.1.5 Interfaces With Other Equipment or Systems**

The on-site AC power system, interfaces primarily with the following:

1. The Offsite AC Power System
2. The On-site DC Power Systems
3. Plant loads requiring SCN and SC3 power.

The SC3 SDGs are electric start and radiator cooled and do not require plant mechanical support services for operation (such as plant auxiliary cooling or instrument air). Either of the SDGs can support all the SC2 and 3 EDS loads and SC1 EDS loads (three divisions) that require power for completion of Safety Category 1, 2, and 3 functions (i.e., either SDG can support active decay heat removal). There is on-site fuel storage in large storage tanks and day tanks for each SDG. The large storage tanks hold enough fuel for at least seven days of full power operation of the SDGs at rated load. The local day tanks are in addition to the seven-day supply of fuel to support online testing; the day tanks alone provide up to eight hours of full power operation at rated load.

The most important loads on the A21 and B21 busses are the SC1 EDS UPS, battery chargers, and Regulating Voltage Transformers (RVTs). Next in line as a matter of importance are the SC2 EDS LGs that provide power to the SC2 UPSs, battery chargers, and RVTs. Along with the two SC2 EDS LGs, the four FMCRD UPSs are of high importance.

#### **8.4.1.6 System and Equipment Operation**

The normal preferred power source to the plant is via the UAT which is connected to the GSU and the Main Generator. The alternate preferred power source is via the RAT, which is directly connected to the switchyard. With the plant online, the MV SCN busses, and therefore the A21 and B21 LV busses, are normally aligned to the normal preferred power source through separate windings of the UAT. With the plant offline, the MV SCN busses and therefore the A21 and B21 LV busses are aligned to the alternate preferred power source through separate windings of the RAT.

In addition to the typical alignments previously described, it is also possible to align the A21 and B21 busses to the alternate preferred power source through separate windings of the RAT during power operation. This could be accomplished manually via operator actions or automatically via an MV fast bus transformer, such as would occur during a faulted UAT scenario. Finally, with the plant offline, the A21 and B21 busses can be aligned to normal preferred power source through separate windings of the UAT with the Main Generator Breaker open in a backfeed alignment through the GSU. All transfers between the normal preferred power source and the alternate preferred power source occur at the MV busses.

The SC2/SC3 EDS provides power to SC1, SC2, SC3, and SCN loads as needed. SC3 SDGs are utilized as needed to supply power to SC1, SC2, SC3, and SCN functions. However, this power is not needed to support plant shutdown or cooldown functions as previously noted.

#### **8.4.1.7 Instrumentation and Control**

The BWRX-300 on-site AC power systems are to be monitored and controlled by DCIS equipment of a corresponding safety classification. Generally, the DCIS control systems are expected to operate and monitor the plant transformers and feeder breakers, including load breakers for pumps, adjustable speed drives, or other plant equipment.



#### **8.4.1.8 Monitoring, Inspection, Testing, and Maintenance**

If electrical equipment is found to be inoperable or failing, operators would be alerted, and an inspection of the equipment would be initiated. operators would make the decision based on inspection to replace or repair or monitor the equipment for any further degradation.

The design of the plant electrical systems accommodates online testing and inspection to the extent practical. Condition monitoring equipment is included where appropriate.

For the SC2 and SC3 EDS, periodic starting and loading of the SDGs is expected to be required by plant technical specifications. For the SCN EDS, there are no technical specification surveillance requirements.

To the extent practical, the design of the plant electrical systems includes the capability to isolate, maintain, and test individual components and equipment while the plant is online.

Nuclear power plant commissioning usually involves three types of testing, in order of plant completion:

- Construction tests
- Pre-operational tests
- Startup/power ascension tests

The first two test types may overlap, but the last test type requires that the first two be completed before fuel loading is allowed. The plant electrical system is to be involved in all testing but note that no significant plant testing can be done until the plant electrical system is operational. In this context, "operational" implies that the system powers mechanical loads that need testing and power the plant DCIS loads needed to monitor, control, and record the plant systems to be tested. This implies that the plant electrical system should be among the first operational installed systems:

- The design on the on-site AC power systems include provisions for connection to construction power.
- The plant electrical system switchgear and distribution equipment include access points in the front, back, and sides as needed to support efficiency in constructability.

Where possible, it is intended to purchase electrical components that do not require maintenance for a 60-year plant lifetime. The large BWRX-300 plant transformers (GSU, UAT, and RAT) have a realistic 30-year lifetime. Condition monitoring of the transformers and other important equipment is provided to assist in determining appropriate maintenance and replacement intervals.

NOTE: The transformers are expected to be conservatively rated to reduce heat and maximize longevity.

The BWRX-300 plant is designed to meet a 60-year design life assuming required maintenance, operational program requirements, and equipment replacements, as needed, are accomplished. Replacement frequencies follow manufacturer guidance and in combination with available monitoring data if available.

#### **8.4.1.9 Radiological Aspects**

The on-site AC power system is qualified for the safety function, SC and corresponding locations as discussed in Chapter 3, Section 3.9, Environmental Qualification.

#### **8.4.1.10 Performance and Safety Evaluation**

The on-site AC electrical power system is expected to perform their design functions in normal and abnormal conditions. The reliability of items important to safety is to be commensurate with its safety significance.

### **8.4.2 On-site Direct Current Power Systems**

#### **8.4.2.1 System and Equipment Functions**

The on-site direct current power system is divided into three subsystems with three safety classes, SC1, SC2 and SCN. The system and equipment functions are described below:

- SC1 Direct Current Power System

The SC1 DC power system powers the SC1 DCIS equipment responsible for automatic shutdown and decay heat removal. The SC1 DCIS safety functions are typically executed in a fail-safe manner through removal of power. The SC1 EDS is a DL3 system limited to only supply power to SC1 SSC.

- SC2 Direct Current Power System

The loads assigned to the SC2 UPSs and batteries are those required for the BWRX-300 to achieve cold shutdown and include the equipment that provides a diverse, active alternative to SC1 vessel injection and vessel and containment heat removal functions. The SC2 DC power system is a DL4a system.

- SCN Direct Current Power System

The SCN DC power system is designed to provide power to SCN DC loads, such as protective relaying and breaker control power.

#### **8.4.2.2 Safety Design Bases**

The safety design basis criteria are provided below for each of the three direct current subsystems.

- SC1 Direct Current Power System

- The design of the SC1 DC system conforms to the applicable requirements given in IEEE Std 308 (Reference 8.8-13) and IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (Reference 8.8-29).
- The SC 1 batteries meet the qualification requirements of IEC/IEEE 60780-323, "IEC/IEEE International Standard - Nuclear facilities -- Electrical equipment important to safety – Qualification" (Reference 8.8-17) and IEC/IEEE 60980-344, "IEEE/IEC International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification" (Reference 8.8-18).
- A single failure in the SC1 EDS that results in a loss of any single divisional AC output shall not result in loss of both AC outputs within that division SC2 Direct Current Power System.
- The SC1 EDS batteries shall have enough stored capacity without chargers to independently supply loads continuously for at least 72 hours with a minimum final discharge voltage sufficient to support the associated loads.
- Each of the SC1 EDS UPSs shall be able to supply 100% of the divisional load.

- The SC1 EDS divisions shall be electrically separate such that no electrical connection exists between a UPS in one SC1 EDS division and a component in another SC1 EDS division.
- SC2 Direct Current Power System
  - The SC2 EDS batteries shall have enough stored capacity without chargers to independently supply loads continuously for at least 72 hours with a minimum final discharge voltage sufficient to support the associated loads
- SCN Direct Current Power System

SCN systems provide no safety function.

#### **8.4.2.3 Description**

The on-site direct current power system description for each of the three subsystems is provided below:

- SC1 Direct Current Power System

Figure 8.8-2, Safety Class 1 Low Voltage Uninterruptible Power Supply Systems represents a typical design of each of the three SC1 EDS divisions. As shown in the main electrical one-line in Figure 8.8-1, and the SC2 and 3 EDS one-line in Figure 8.8.2, the SC2 and 3 EDS provides separate feeds from an SC3 LV bus to the three divisions of the SC1 EDS equipment. Each division can be continuously energized from multiple sources: offsite power (normal supply), standby diesel generator power (backup supply), portable generator power (external building connections), or individual division battery power (emergency supply). Independent of offsite or diesel AC power, the SC1 batteries in the SC1 EDS provide backup power for at least 72 hours for monitoring and mitigation loads required following a loss of AC power event. The 72-hour requirement is consistent with the overall plant design basis and regulatory requirements to preclude the need for AC power for 72 hours, while mitigating the effect of a DBA. The external connections provide for connection of EME/FLEX as needed.

Each SC1 division has its own battery and two redundant UPSs. The two UPSs in each division are primarily for the SC1 DCIS system to provide redundant divisional power feeds. Loss of any individual divisional UPS still allows the DCIS of that division to function. Therefore, reliability is enhanced with the redundant UPSs that provide power to each division of the DCIS. In addition, all UPSs normally operate directly from AC power, allowing the batteries to be fully tested without losing the divisional outputs (albeit the 72-hour coping time is unavailable from the battery in test during that time).

While each SC1 divisional electrical system uses two redundant UPSs, the RVT is de-energized in the normal system lineup and must be manually placed into service. This means automatic swap over to the UPS bypass AC source is not provided, as required by IAEA SSG-34, "Design of Electrical Power Systems for Nuclear Power Plants" (Reference 8.8-22) and the guidance of REGDOC-2.5.2 (Reference 8.8-8). The reason for operating without automatic swap over to the bypass power is to mitigate the Forsmark event (U.S. NRC ML102070360 (Reference 8.8-10)). In that event, electrical transients starting in the transmission system traveled through the plant electrical system and consequently disabled the SC1 inverters. The BWRX-300 battery chargers, UPSs, and RVTs are hardened against such an event with TVSSs on the AC input lines. While the TVSSs are designed to prevent HV spikes from being transferred through the EDS, transients could

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

still disable the UPS. The intent of an automatic swap over to the UPS bypass RVT is to maintain the required divisional output should the UPS inverter fail. With the BWRX-300 design, the second UPS in each division serves the same purpose as the automatic swap over to an RVT. However, the second UPS comes with the added benefit that it is already in service, and it is fed from a different power source.

Each SC1 UPS has a separate output bus (distribution panel) for distribution of AC power. Use of a Manual Transfer Switch (MTS) in the UPS output circuit isolates the UPS output from the loads when the RVT is placed in service. In this configuration, the RVT directly powers the respective UPS distribution panel and maintenance or testing of the UPS can be done online without disturbing UPS loads. The AC power supplied through the UPS bypass RVT would be powered from either the SDG backed busses or the portable generator connections.

There are two battery chargers per SC1 division. Each battery charger has the capacity to power all safety functions for the division. The redundant battery chargers and bus configuration are arranged to provide operational flexibility to the system.

The two UPSs per SC1 division provide a single-phase 230 VAC output. The higher voltage was chosen to reduce voltage drops between the UPSs and SC1 loads. The batteries produce 250 VDC (nominal) power connected to a DC distribution bus in each division. The DC distribution bus provides the DC input to the UPSs and to remote DC power panels where needed.

Each SC1 UPS distribution panel feeds multiple power panels to provide redundancy from each division. The power panels ultimately feed the individual SC1 loads. Where required, the UPSs are protected at the individual load level with current limiting fuses such that the resulting fault currents are within the transient rated current capability of the inverter. Individual faults have no effect on other loads on the same power panel. All SC1 DCIS components are redundantly powered from the SC1 EDS divisional UPSs. Loads with the ability to accept only a single supply supplied by a single feed in line with an automatic transfer switch with input separately derived from the two UPS. Therefore, failure of one divisional UPS has no adverse effect on plant operations or safety.

Each of the SC1 divisional electrical systems provide DC power and uninterruptible AC power where needed for SC1 loads. Most SC1 loads are located throughout the RB but can also be located in other areas of the plant like the MCR in the CB.

All SC1 EDS batteries and electrical switchgear components in the RB are in separated rooms that are fire and flood-barriered from each other by division. The SC1 components and equipment in these areas are qualified to environmental and seismic conditions such that no active equipment is needed to maintain or support normal operation or credited post AOO or accident coping period.

The Division 1 and Division 2 electrical system configurations of the SC1 EDS are identical. The Division 3 electrical system configuration has slightly less load. A small selection of loads is allocated only to Division 1 and 2 as described in Chapter 7, Instrumentation and Control. Each division is designed to the maximum load and mission time of the three divisions

- SC2 Direct Current Power System

Figure 8.8-3, Safety Class 3 Low Voltage Busses A21 and B21, depicts the SC3 A21 and B21 LV bus configurations and typical SC2 and 3 EDS loads.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 8.8-4, Safety Class 2 Low Voltage Uninterruptible Power Supply Systems, depicts the SC2 EDS MCCs for LG A and LG B, with the battery and battery charger configuration.

Figure 8.8-5, Safety Class 2 Low Voltage Uninterruptible Power Supply Distribution, depicts the SC2 EDS UPS distribution configuration for LG A and LG.

The SC2 direct current power system includes two separate load groups (A and B), and four separate UPSs for the FMCRDs. The SC2 load groups A and B are arranged similar to the SC1 Divisions with the exception that each load group contains only a single UPS verses the two per SC1 division. Each load group is independently powered from the associated SC3 AC train A or B. The FMCRD SC2 UPSs are arranged such that two are powered from the SC3 A21 LV bus and two are powered from the SC3 B21 LV bus.

The SC2 LV LGs include surge protection via a transient voltage surge suppressor to prevent HV spikes propagating to the battery chargers, UPSs, and RVTs. Each LG has an individual UPS in support of the respective SC2 DCIS division, along with other loads. The LG UPS provide three-phase 208/120 VAC outputs. Redundant circuits are provided from separate UPSs to support redundant power sources to the DCIS. The SC2 UPSs are the AC power supplies associated with LGs A and B. The LG A MCC is powered from the SC3 A21 LV bus, and the LG B MCC is powered from the SC3 B21 LV bus.

- SCN Direct Current Power System

Figure 8.8-8, SCN 125 VDC System, represents the design of the SCN 125 VDC System. The SCN DC system is designed to provide power to SCN DC loads, such as protective relaying and breaker control power. As shown in Figure 8.8-8, two independent 125 VDC batteries are provided, supplying power to two independent 125 VDC busses. The redundant configuration is compliant with NERC requirements for separate backup power for Bulk Power Systems. A battery mission time of eight hours is selected to align with typical designs for Bulk Power Systems substations. Each DC bus and each battery is normally supplied power and charging current from its associated charger. A standby charger capable of being aligned to either bus or battery is provided in case any given charger is out of service. Each charger is designed to supply 100% of the required load of a single bus, while charging the battery. The two primary chargers derive LV AC power from separate 600 VAC SCN busses. The standby charger is supplied through a transfer switch, which can derive power from either the A21 or B21 SDG backed busses. The standby charger connection to the SDG busses provides operational flexibility in the event of an extended bus outage.

Figure 8.8-9, SCN UPS 208/120 VAC System represents the design of the SCN 208/120 VAC system. The SCN UPS system is designed to provide power to SCN loads. Two independent UPSs are provided that provide a three-phase 208/120 VAC output to independent busses. The UPSs normally derive power from their associated 125 VDC bus. The UPSs are equipped with an MTS aligned to an RVT. The RVTs derive power from a separate AC source than that from which the associated battery charger is supplied.

The following design requirements are applicable to all direct current power systems described above:

1. All EDS batteries and equipment are physically separated and electrically independent in accordance with the requirements given in IEC 60709 (Reference 8.8-15) and IEEE Std 384 (Reference 8.8-30).
2. All battery chargers and UPSs disconnect from the AC input whenever that input varies outside the range of +/-15% voltage from nominal for greater than 60 seconds.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

3. All battery chargers are able to function as battery eliminators.
4. All battery chargers produce a filtered DC output that meets the requirements of IEC 61000-4, "Electromagnetic Compatibility Testing and Measurement Package" (Reference 8.8-31).
5. All battery chargers are housed NEMA Type 1 ventilated enclosures.
6. All battery chargers include overvoltage protection on the output that shuts down the charger upon actuation.
7. All battery chargers are of a current limiting design.
8. All battery banks are designed to permit replacement of individual cells. The plant DC systems consist of an ungrounded system with ground detection.
9. All UPSs maintain rated output voltage  $\pm 5\%$  from zero to rated load.
10. All UPSs are sized to provide the current of a single faulted load in addition to all other normal loads, while still providing normal output voltage.
11. The UPS bypass RVTs maintain a nominal output voltage within  $\pm 5\%$  over an input voltage variation of  $+10\%$  and  $-15\%$ .
12. The battery chargers, UPS, lighting, and convenience power all work continuously without operator intervention.
13. Category B, Transient voltage surge suppressors are used on the input circuit to battery chargers, UPSs, and associated bypass circuits to protect against voltage surges.
14. In accordance with IEEE 1050-2004, adequate filtering is provided by the battery charger, UPS, and regulating transformers to ensure that propagation of EMI/RFI of the levels allowed by IEC 61000-3 (Reference 8.8-32) does not adversely affect other plant equipment.

The following design requirements are applicable to the SC1 DC subsystem:

1. The SC1 DC system meets the recommended practices in IEEE 946-2020 (Reference 8.8-14) for design of DC power systems for stationary applications.
2. SC1 batteries are designed with consideration of the recommended practices in IEEE 484-2019 (Reference 8.8-16) for installation design and installation of vented lead-acid batteries for stationary applications.
3. Maintenance, testing, and replacement activities for the SC1 EDS batteries consider the recommended practices in IEEE 450-2020.
4. The SC1 batteries are sized with an expected service life of 20 years using the methodology given in IEEE 485-2020 (Reference 8.8-20).
5. Each of the SC1 EDS UPSs can supply 100% of the divisional load.
6. The SC1 Divisions are electrically separate such that no electrical connection exists between the DC batteries and UPS in one division and a component in another division.
7. The SC1 UPS outputs provide a single-phase nominal AC voltage of 230 VAC, 60 Hz.

8. The SC1 battery banks are designed to supply DC power at 250 VDC nominal.
9. The SC1 EDS provides battery-backed power to support any accident monitoring and mitigation functions that are a part of DL3. In the event that normal AC power is lost, the SC1 EDS continues to provide uninterruptible power for a minimum mission time of 72 hours.

The following design requirements are applicable to the SC2 DC subsystem:

1. Any SC2 load requiring power for completion of a Safety Category 2 function is powered from the SC2 EDS UPSs or batteries and cannot be powered directly from the SC3 busses. The SC2 and 3 EDS are designed with redundancy such that the failure of a single electrical bus or MCC has no adverse effect on the redundant counterpart.
2. Redundant SC2 battery chargers support equalization and battery testing without impact to downstream loads
3. The SC2 battery banks are designed to supply DC power at 250 VDC nominal
4. The SC2 DC and UPS systems are powered from diesel-backed sources

The following design requirements are applicable to the SCN DC subsystem:

1. The SCN EDS UPS outputs provides a three-phase nominal AC voltage of 208/120 VAC, 60 Hz.
2. The SCN EDS battery banks are designed to supply DC power at 125 VDC nominal.
3. The SCN EDS DC systems supply power to the inverters that are part of the SCN UPS systems.
4. The SCN EDS batteries have enough stored capacity without chargers to independently supply loads continuously for at least eight hours, with a minimum final discharge voltage sufficient to support the associated loads.

#### **8.4.2.4 Materials**

Materials associated with the SC of the individual systems. SSC are evaluated based on the SC and functions. Materials are evaluated and selected based on the environmental conditions and SC of the equipment as discussed in PSAR Chapter 18.

#### **8.4.2.5 Interfaces With Other Equipment or Systems**

The three DC subsystems supply loads of equivalent SC. Input AC power for such systems is derived from AC sources as identified in the figures included in this chapter. The systems also include interface with the DCIS of equivalent SC for control and monitoring.

#### **8.4.2.6 System and Equipment Operation**

The DC subsystems are designed to perform their associated function without operator interaction. Upon loss of AC power, the systems will automatically transition to being supplied by the associated battery.

#### **8.4.2.7 Instrumentation and Control**

The BWRX-300 electrical systems are to be monitored and controlled by DCIS equipment of a corresponding safety classification. Generally, the DCIS control systems are expected to operate and monitor the plant transformers and feeder breakers, including load breakers for pumps, adjustable speed drives, or other plant equipment.

MV breakers are controlled by protective relays established by the designer to protect the electrical system components. For example, the condensate pump motor breaker includes protective relays that monitor overcurrent, balanced phases, undervoltage, underfrequency, and ground and phase faults. The protective relay trips the breaker, and a trip command has precedence over a close command. The plant DCIS can command a breaker to open or close, but a DCIS close command can be overridden by the protective relay. Either a DCIS trip or protective relay trip command opens the breaker.

The protective relays are interfaced to the plant DCIS and can provide multiple signals to the attached plant controller. Potential signals include:

- Phase voltage
- Phase current
- Power factor
- Power
- Apparent power
- Reactive power
- Total running time
- Total off time
- Number of starts

NOTE: These signals are provided “automatically” by the protective relays and no further instrumentation is needed.

Although modern microprocessor based protective relays are considerably more flexible, cost effective, and accurate than older analog designs (that cannot provide equivalent protection), the relays represent a separate DCIS network. The relays are considered critical digital assets in the U.S. and require appropriate cyber security monitoring and physical access protection.

For components and equipment in the plant electrical systems that are monitored by the DCIS, status inputs and alarm indications shall be provided to the plant DCIS.

Refer to Chapter 7 for a detailed discussion of the BWRX-300 I&C System design and operation.

#### **8.4.2.8 Monitoring, Inspection, Testing, and Maintenance**

Condition monitoring equipment is included where appropriate.

The SC1, SC2 and SCN batteries require periodic testing under load to verify capacity. Although the loads are expected to be out of service, the associated DCIS loads can remain powered because the UPSs have bypass AC power available.

The design of the plant electrical systems accommodates online testing and inspection to the extent practical. Condition monitoring equipment is included where appropriate. For the SC1 EDS, the UPSs are in pairs per division, and testing can be done online without losing the complete divisional UPS output.

Required electrical system testing is governed by the plant technical specification and industry standards. The electrical systems are designed to allow required tests to be performed with minimal disturbance to the balance of the system. Specific steps and alignments necessary for the successful performance of testing are to be controlled by plant procedures.



All maintenance can be done during outages. However, given the redundancy in the electrical system, most maintenance can also be done with the plant online and with only a small power reduction. All load centers necessary for power generation can be replaced online, although some replacements may require a plant power reduction.

While the expected service life of the plant batteries is 20 years, occasionally a battery cell may fail. These issues are to be appropriately alarmed by the battery monitoring systems. With optimum environmental conditions (e.g., temperature and humidity), limiting the discharge occasions to periodic surveillance and preventive maintenance testing, and having constant condition monitoring for cell degradation, a 30-year service life of the plant batteries is also possible. All batteries can be replaced either online or during outages.

Depending on the battery type, a battery may periodically need a maintenance equalizing charge that raises battery voltage to levels that can cause unnecessary stress on battery loads. With the redundant battery charger design configuration, one charger can supply the equalizing charge to the battery and be isolated from the DC loads while the other charger carries the DC loads at normal voltages.

#### **8.4.2.9 Radiological Aspects**

The electrical system is expected and designed to function as specified by its SC. The safety function of the electrical SSC must be accomplished for safety for a facility or activity to prevent or to mitigate radiological consequences of normal operation, AOOs and accident conditions.

#### **8.4.2.10 Performance and Safety Evaluation**

The on-site DC power system are expected to perform their design functions in normal and abnormal conditions.

Electrical power is supplied to the I&C systems. Refer to Chapter 7 for a further discussion.

### **8.5 Electrical Cables and Raceways**

#### **8.5.1 System and Equipment Functions**

The cabling for the BWRX-300 provides electrical power to designated loads. Fiber cabling is used for signal transmissions. SC loads are provided appropriate SC power from the EDS. Electrical cables and raceways are passive equipment and do not perform and active functions.

#### **8.5.2 Safety Design Bases**

The cabling and raceways are designed to meet the following SC design requirements:

1. The plant SC electrical systems are designed to meet the applicable requirements of IEC 60709 (Reference 8.8-15) for cabling and system separation and independence of SC systems. Power and signal/control cables will be separated per the guidance provided in IEC 60709 (Reference 8.8-15).
2. Separation for SCN components and systems are derived from the application of IEC 60709 (Reference 8.8-15) as appropriate.
3. Current-carrying cables in the plant electrical systems are qualified to meet the IEEE Std 383 (Reference 8.8-21) requirements for cables and splices for nuclear facilities.

The cables and raceways are designed commensurate with the SC of the loads and functions being provided.

### **8.5.3 Description**

Separation of cables by divisions or load groups are provided by routing in raceways that provide the separation by distance and routing in which the distance is maintained to ensure that the designated SC is maintained. The raceways provide support for the cables commensurate with the SC designations of the individual SC of the cables. Seismic and EQ considerations are also factored into the design of the raceways for proper support and safety functions. Penetrations are designed based on SC considerations.

### **8.5.4 Materials**

Materials are associated with the SC of the individual systems. Materials are evaluated and selected based on the environmental conditions to which the cables/raceways are exposed in conjunction with the SC of the equipment being served. SC of the equipment is discussed in Chapter 3, Section 3.9, Environmental Qualification.

### **8.5.5 Interfaces With Other Equipment or Systems**

Cabling and raceways are passive components that provide power from electrical supplies to components and/or loads.

Electrical cables and raceways are permanently installed passive components to support operation of connected loads.

### **8.5.6 Instrumentation and Control**

While electrical cables transmit I&C signals throughout the plant, there are no instrumentation and control features specific to the electrical cabling and raceway systems.

### **8.5.7 Monitoring, Inspection, Testing, and Maintenance**

Normally the cabling and raceways are designed to not require any inspections once they have passed installation and commissioning testing. However, cabling and raceways are inspected based on operations surveillance programs, maintenance and testing programs during normal operations and outages as appropriate.

Cabling and raceways testing is based on preventive and predictive maintenance programs. Site testing requirements will also be used for post installation testing and inspection.

### **8.5.1 Radiological Aspects**

The cables and raceways are designed for their location to maintain their required function. The cables and raceways are qualified for the anticipated radiological environment expected during the lifetime of the plant.

### **8.5.2 Performance and Safety Evaluation**

The cables and raceways are designed to perform required functions during all plant modes and under all anticipated normal and abnormal conditions.

## **8.6 Grounding, Lightning Protection and Electromagnetic Compatibility**

### **8.6.1 System and Equipment Functions**

The plant grounding system is part of the SCN EDS. Grounding is important for both safety of plant personnel and electrical system protection, and it is critical to the operability and performance of the BWRX-300 DCIS.

As with any grounding system, the overall function of the plant grounding networks is to protect personnel and equipment from induced voltages and currents by providing a low impedance path to the earth.

### **8.6.2 Safety Design Basis**

The overall grounding system consists of a buried grounding grid and includes different types of grounding systems that all connect to a common plant grounding grid.

Building grounds (connections of the building foundations and external structures to the grounding grid) are provided. Equipment grounds (connection of equipment to the internal grounding system of the buildings) are provided for equipment protection. Instrument grounds (separate grounding connection to the grounding grid for analog and digital instrumentation systems) are provided for instrumentation protection and operation. The lightning protection system provides protection for equipment against lightning induced surges.

### **8.6.3 Description**

The electrical grounding system is comprised of the following grounding networks connected to a plant grounding grid:

1. An instrument and computer grounding network
2. An equipment-grounding network for grounding electrical equipment (e.g., transformer, switchgear, motors, distribution panels, cables) and selected mechanical components (for example, fuel tanks, chemical tanks)
3. Building grounds
4. A lightning protection network for protection of all structures, transformers, and equipment

The instrumentation grounding system is provided to include both the plant analog (relays, solenoids, etc.) and digital instrumentation systems. The plant instrumentation is grounded through a separate insulated radial grounding system composed of busses and insulated cables. The instrumentation grounding system is connected to a discrete point of the station grounding grid at a dedicated instrumentation grounding rod by exothermic welding. The instrumentation grounding system is insulated from all other grounding and surge protection circuits up to the point of connection at the ground grid. It is recognized that there are numerous accepted grounding techniques, and that the actual installation of a ground system is made with reference to the recommendations of the I&C equipment manufacturers since the techniques used to solve one problem may result in the creation of a different problem (IEEE Std 1050 (Reference 8.8-25)).

The equipment-grounding network is such that all major equipment, structures, and tanks are grounded with two diagonally opposite ground connections. The ground bus of all switchgear assemblies, motor control centers and control cabinets are connected to the station ground grid through at least two parallel paths. Bare copper risers are furnished for all underground electrical ducts and equipment, and for connections to the grounding systems within buildings. One bare copper cable is installed with each underground electrical duct run, and all metallic hardware in each manhole is connected to the cable.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

A plant grounding grid consisting of bare copper cables is provided to limit step and touch potentials to safe values under all fault conditions. The buried grid is connected to the ground mat at the switchyard and connected to systems within the buildings by a bare copper loop, which encircles each building.

The plant's main generator is grounded with a neutral grounding device to limit the magnitude of fault current due to a solid phase-to-ground fault. The impedance of the neutral grounding device limits the maximum phase-to-ground current under short-circuit conditions.

The on-site, MV AC distribution system is resistance grounded at the neutral point of the LV windings of the UAT and RAT. The neutral point of the generator windings of the standby on-site AC power supply is through neutral resistors, sized for continuous operation in the event of a ground fault.

The neutral point of the LV AC distribution systems is solidly grounded to ensure proper coordination of ground fault protection. The DC systems are ungrounded.

The lightning protection system covers all major plant structures and is designed to prevent direct lightning strikes to the buildings, electric power equipment and instruments. It consists of air terminals, bare downcomers, and buried grounding electrodes. Lightning arresters are provided for each phase of all tie lines connecting the plant electrical systems to the switchyard and offsite lines. These arresters are connected to the high-voltage terminals of the main step up transformers, UAT, and RAT. Plant instrumentation located outdoors or connected to cabling running outdoors is provided with surge suppression devices to protect the equipment from lightning induced surges.

The plant grounding system is part of the SCN EDS. Grounding is important for both safety of plant personnel and electrical system protection, and it is critical to the operability and performance of the BWRX-300 DCIS.

System Requirements for the plant grounding system contain the following:

1. The SCN EDS includes an electrical grounding and lightning protection system with a plant grounding grid.
2. The plant grounding grid meets the applicable requirements of IEEE Std 80, "IEEE Guide for Safety in AC Substation Grounding" (Reference 8.8-1), IEEE Std 81, "IEEE Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System" (Reference 8.8-2), and IEEE 142-2007.
3. The electrical grounding protection system includes an instrument and computer grounding network that is separate from the standard equipment and system ground network.
4. Lightning and surge suppression meets the applicable requirements given in IEEE Std C62.23, "IEEE Application Guide for Surge Protection of Electric Generating Plants" (Reference 8.8-6), IEEE Std 666, "Design Guide for Electric Power Service Systems for Generating Stations" (Reference 8.8-7), IEEE Std 1050 (Reference 8.8-4), IEEE Std 665, "IEEE Guide for Generation Station Grounding" (Reference 8.8-3).

No regulatory guidance is provided for the grounding protection system. Lightning protection is provided in accordance with RG 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants" (Reference 8.8-23). Both systems are designed and installed to the applicable sections of the following codes and standards.

- IEEE Std 80 (Reference 8.8-1)
- IEEE Std 81 (Reference 8.8-2)
- IEEE Std 665 (Reference 8.8-3)
- IEEE Std 1050 (Reference 8.8-4)
- NFPA 780, "Standard for the Installation of Lightning Protection Systems" (Reference 8.8-5)
- IEEE Std C62.23 (Reference 8.8-6)
- IEEE Std 666 (Reference 8.8-7)
- IEC 62305-1, "Protection Against Lightning – Part 1: General Principles" (Reference 8.8-25)

#### **8.6.4 Materials**

A plant grounding grid consisting of bare copper cables is provided to limit step and touch potentials to safe values under all fault conditions. The buried grid is connected to the ground mat at the switchyard and connected to systems within the buildings by a bare copper loop, which encircles each building.

#### **8.6.5 Interfaces With Other Equipment or Systems**

The plant grounding and lightening arresting system is designed to provide protection in the BWRX-300 plant. The grounding system interfaces with all site buildings and structures, all electrical and I&C equipment, and select mechanical equipment needing grounding protection (for example, tanks).

#### **8.6.6 System and Equipment Operation**

The plant grounding and lightening arresting system is designed to provide protection with all equipment and structures in the BWRX-300 plant. As such the system is designed to be passive and does not perform an active safety function. The neutral point of the LV AC distribution systems is solidly grounded to ensure proper coordination of ground fault protection. The DC systems are ungrounded.

#### **8.6.7 Instrumentation and Control**

There are no I&C features required for or applicable to the plant grounding and lightning protection systems.

#### **8.6.8 Monitoring, Inspection, Testing, and Maintenance**

There is no monitoring associated with the plant grounding and lightning protection systems.

Normally the grounding, lightening systems are designed to not require any inspections once they have passed installation and commissioning testing

Much of the grounding systems are underground and inaccessible once installed. Normally the grounding wiring is designed to not require any inspections once they have passed installation and commissioning testing.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

Above ground inspections are conducted based on manufacturer recommendations and industry standard guidance.

There is no testing after installation/construction.

Maintenance of the grounding system is not anticipated after installation/construction.

**8.6.9 Radiological Aspects**

The grounding, lightening protection, and electromagnetic compatibility systems that are exposed to high radiation areas and are designed for the radiological conditions to which they may be exposed.

**8.6.10 Performance and Safety Evaluation**

The electrical grounding system is expected to perform its design functions in normal and abnormal conditions. These systems are completely passive and perform no active functions.

## **8.7 Main Equipment Types**

### **8.7.1 System and Equipment Functions**

The EDS architecture is a configuration of major equipment (MODs, GSU, UAT, RAT, Main Gen. Bkr., Isophase Bus Duct, MV Switchgear, LV Switchgear, battery chargers, batteries, RVTs, UPSs). Refer to Subsections 8.4.1 and 8.4.2 for a discussion of their functions that have been previously discussed.

## **8.8 References**

- 8.8-1 IEEE Std 80, "IEEE Guide for Safety in AC Substation Grounding," Institute of Electrical and Electronics Engineers.
- 8.8-2 IEEE Std 81, "IEEE Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System," Institute of Electrical and Electronics Engineers.
- 8.8-3 IEEE Std 665, "IEEE Guide for Generation Station Grounding," Institute of Electrical and Electronics Engineers.
- 8.8-4 IEEE Std 1050, "IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations," Institute of Electrical and Electronics Engineers.
- 8.8-5 NFPA 780, "Standard for the Installation of Lightning Protection Systems," National Fire Protection Association.
- 8.8-6 IEEE Std C62.23, "IEEE Application Guide for Surge Protection of Electric Generating Plants," Institute of Electrical and Electronics Engineers.
- 8.8-7 IEEE Std 666, "Design Guide for Electric Power Service Systems for Generating Stations," Institute of Electrical and Electronics Engineers.
- 8.8-8 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 8.8-9 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 8.8-10 U.S. NRC ML102070360, "Lessons from Forsmark Electrical Event," U.S. Nuclear Regulatory Commission, November 2010.
- 8.8-11 IEEE Std C37.30.1, "IEEE Standard Requirements for AC High-Voltage Air Switches Rated Above 1000 V," Institute of Electrical and Electronics Engineers.
- 8.8-12 IEEE Std C37.010, "IEEE Application Guide for AC High-Voltage Circuit Breakers > 1000 VAC Rated on a Symmetrical Current Basis," Institute of Electrical and Electronics Engineers.
- 8.8-13 IEEE Std 308, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 8.8-14 IEEE Std 946, "IEEE Recommended Practice for the Design of DC Power Systems for Stationary Applications," Institute of Electrical and Electronics Engineers.
- 8.8-15 IEC 60709, "Nuclear power plants - Instrumentation, control, and electrical power systems important to safety – Separation," International Electrotechnical Commission.
- 8.8-16 IEEE Std 484, "IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronics Engineers.
- 8.8-17 IEC/IEEE 60780-323, "IEC/IEEE International Standard - Nuclear facilities -- Electrical equipment important to safety – Qualification," Institute of Electrical and Electronics Engineers.
- 8.8-18 IEC/IEEE 60980-344, "IEEE/IEC International Standard - Nuclear facilities - Equipment important to safety - Seismic qualification," Institute of Electrical and Electronics Engineers.



NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

- 8.8-19 IEEE Std 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronics Engineers.
- 8.8-20 IEEE Std 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronics Engineers.
- 8.8-21 IEEE Std 383, "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities," Institute of Electrical and Electronics Engineers.
- 8.8-22 IAEA SSG-34, "Design of Electrical Power Systems for Nuclear Power Plants," International Atomic Energy Agency.
- 8.8-23 RG 1.204, "Guidelines for Lightning Protection of Nuclear Power Plants," U.S. Nuclear Regulatory Commission.
- 8.8-24 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."
- 8.8-25 IEC 62305-1, "Protection Against Lightning – Part 1: General Principles," International Electrotechnical Commission.
- 8.8-26 IEEE Std 242, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)," Institute of Electrical and Electronics Engineers.
- 8.8-27 IEEE Std 741, "IEEE Standard for Criteria for the protection of Class 1E Power Systems and Equipment in the Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 8.8-28 IEEE Std C37.06, "IEEE Standard for AC High-Voltage Circuit Breakers Rated on a Symmetrical Current Basis – Preferred Ratings and Related Required Capabilities for Voltages Above 1000V," Institute of Electrical and Electronics Engineers.
- 8.8-29 IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.
- 8.8-30 IEEE Std 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers.
- 8.8-31 IEC 61000-4, "Electromagnetic Compatibility Testing and Measurement Package," International Electrotechnical Commission.
- 8.8-32 IEC 61000-3, "Electromagnetic compatibility (EMC) – Part 3: Limit – All Parts," International Electrotechnical Commission.

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

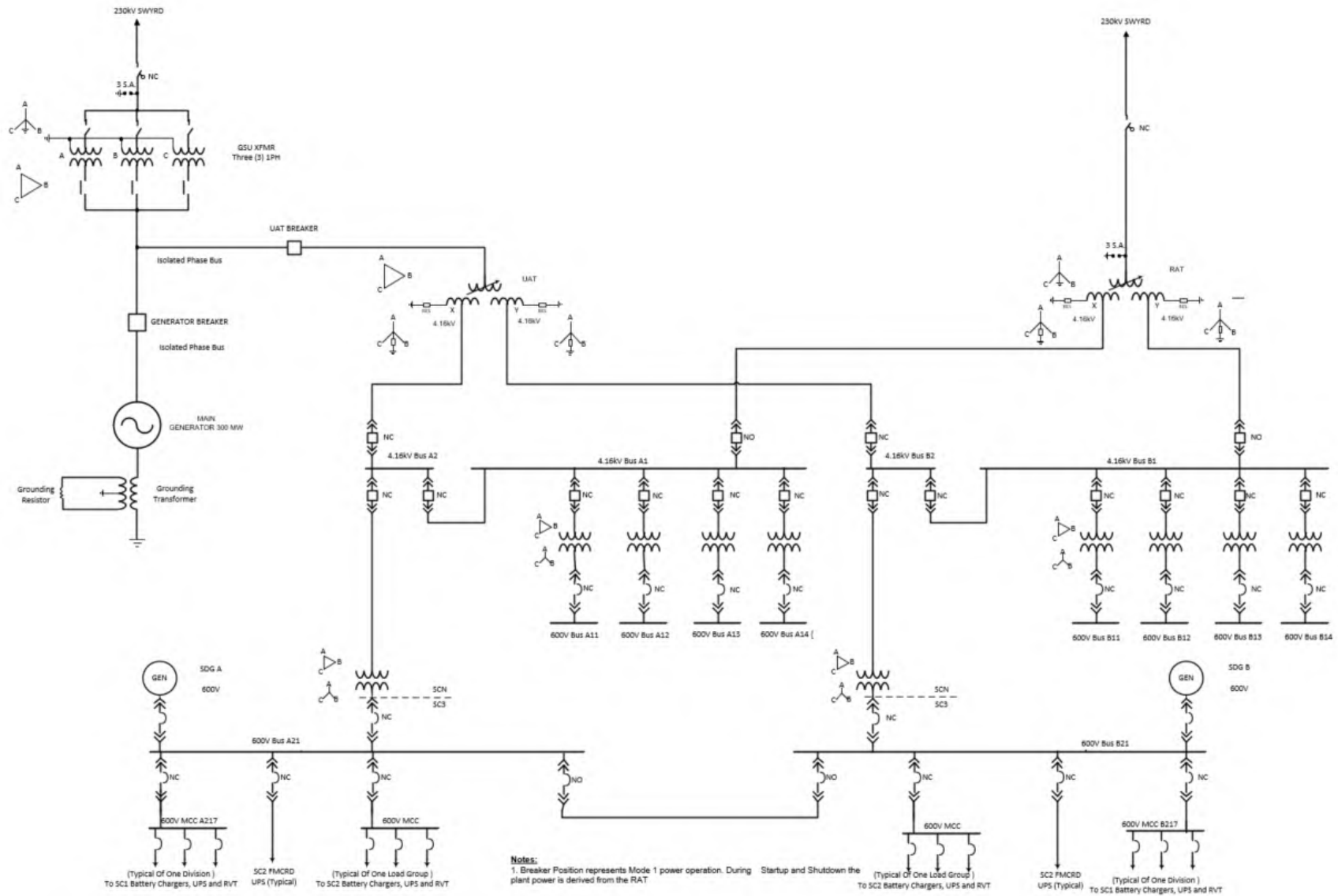


Figure 8.8-1: Main Electrical One-Line

## SC1 LV Division 1 UPS System

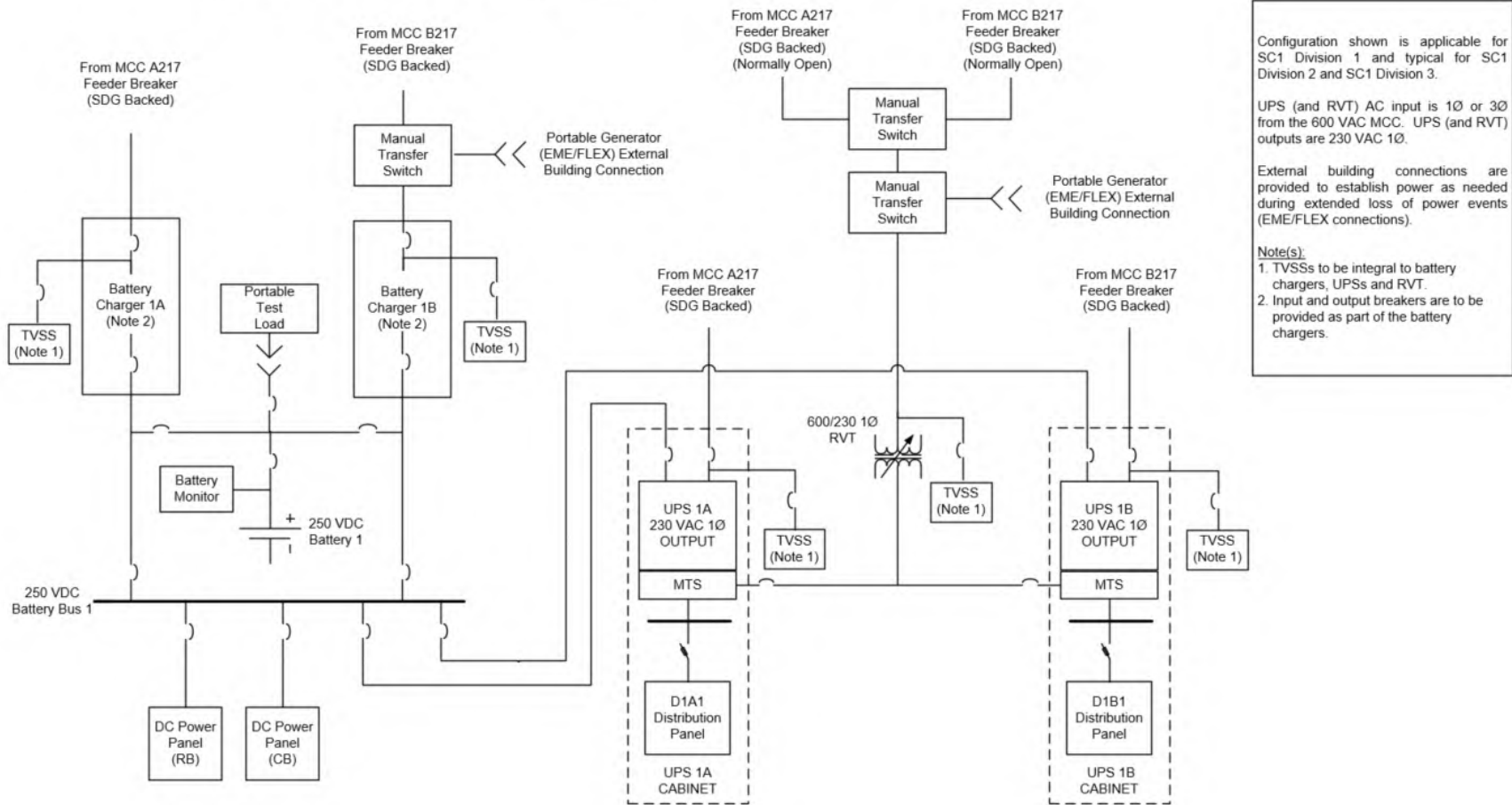


Figure 8.8-2: Safety Class 1 Low Voltage Uninterruptible Power Supply Systems

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

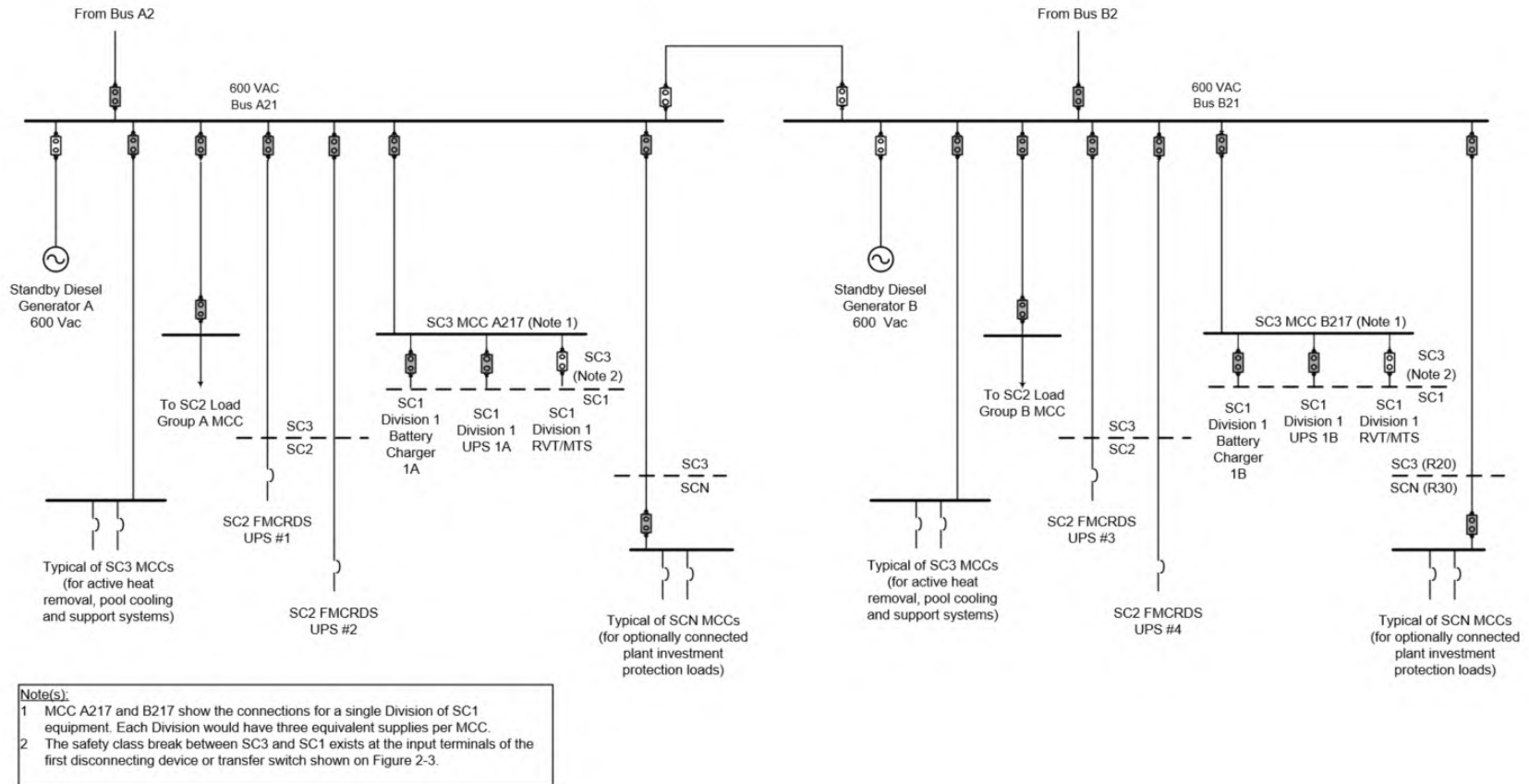


Figure 8.8-3: Safety Class 3 Low Voltage Busses A21 and B21

## SC2 LV LG A UPS System

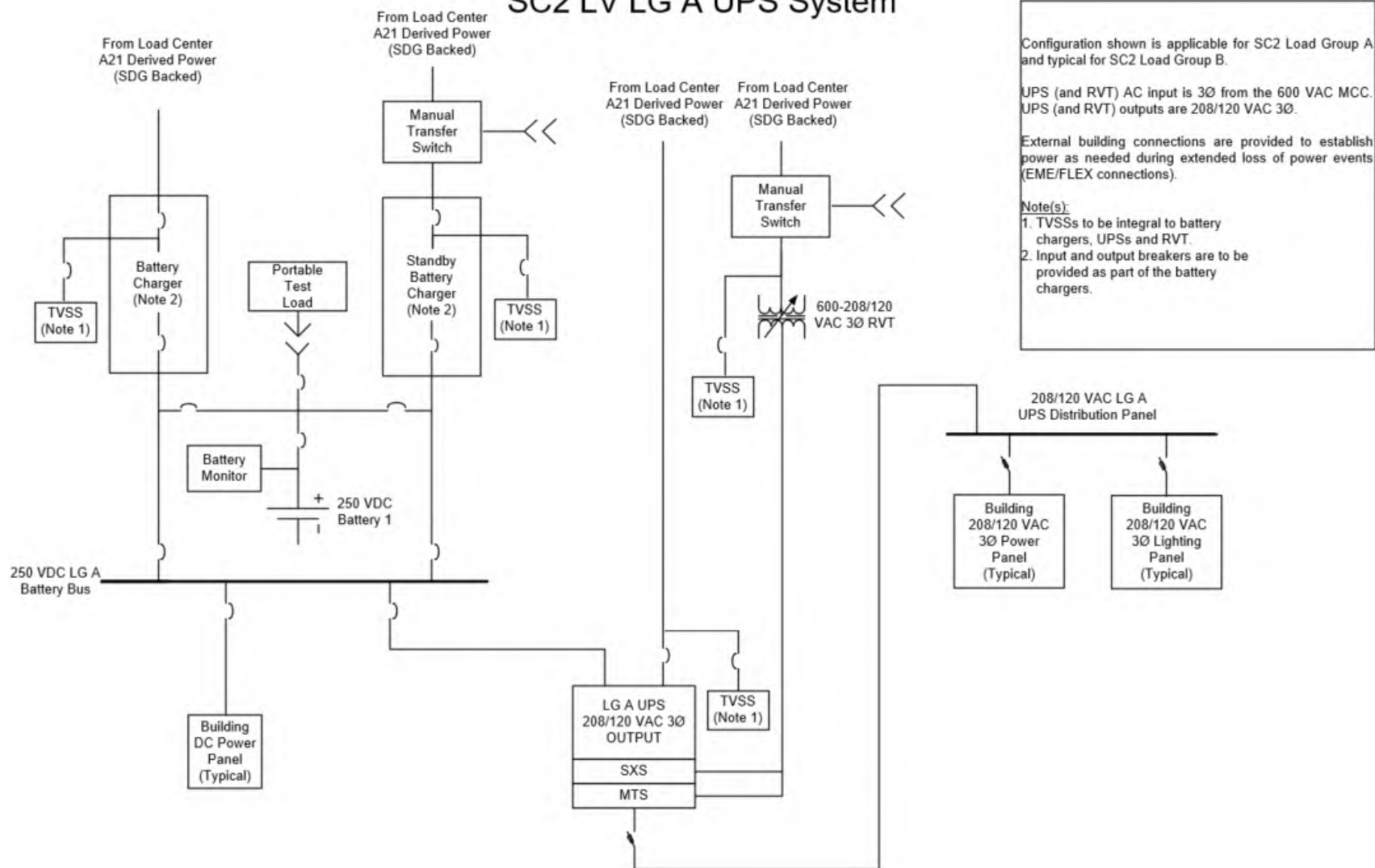


Figure 8.8-4: Safety Class 2 Low Voltage Uninterruptible Power Supply Systems

NEDO-33957 REVISION 0  
NON-PROPRIETARY INFORMATION

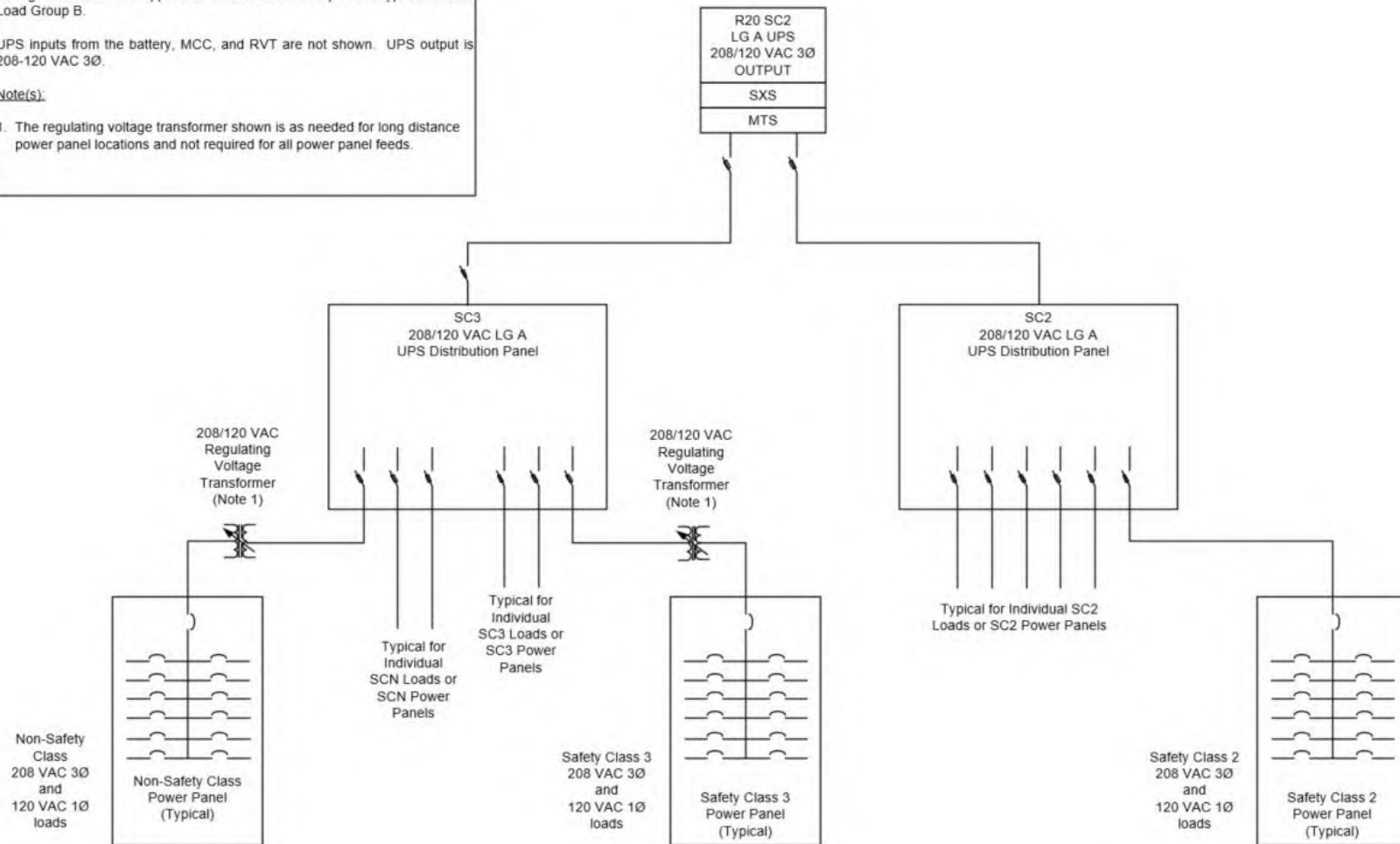
# SC2 LV LG A UPS Distribution

Configuration shown is applicable for SC2 Load Group A and typical for SC2 Load Group B.

UPS inputs from the battery, MCC, and RVT are not shown. UPS output is 208-120 VAC 3Ø.

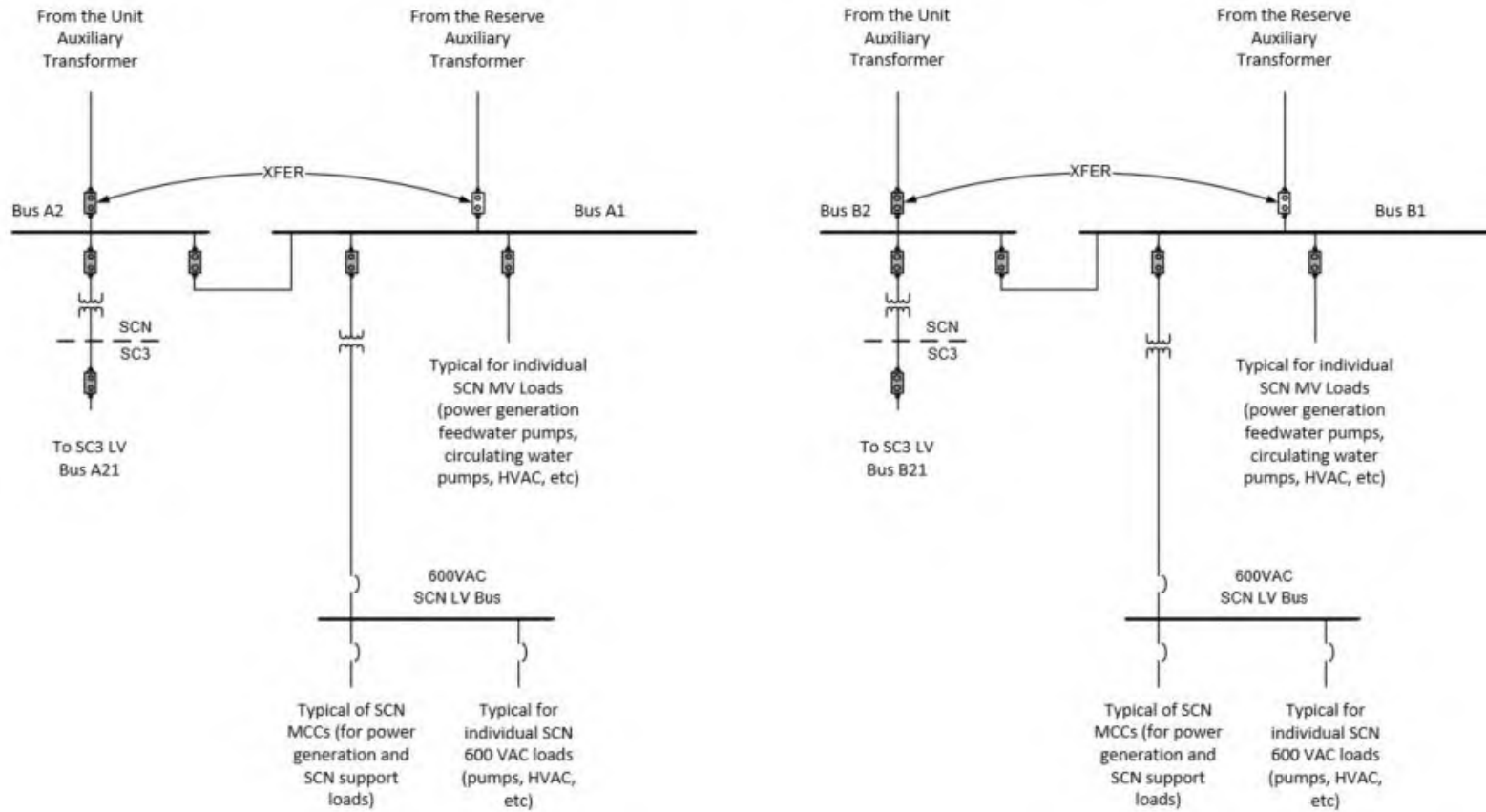
Note(s):

1. The regulating voltage transformer shown is as needed for long distance power panel locations and not required for all power panel feeds.



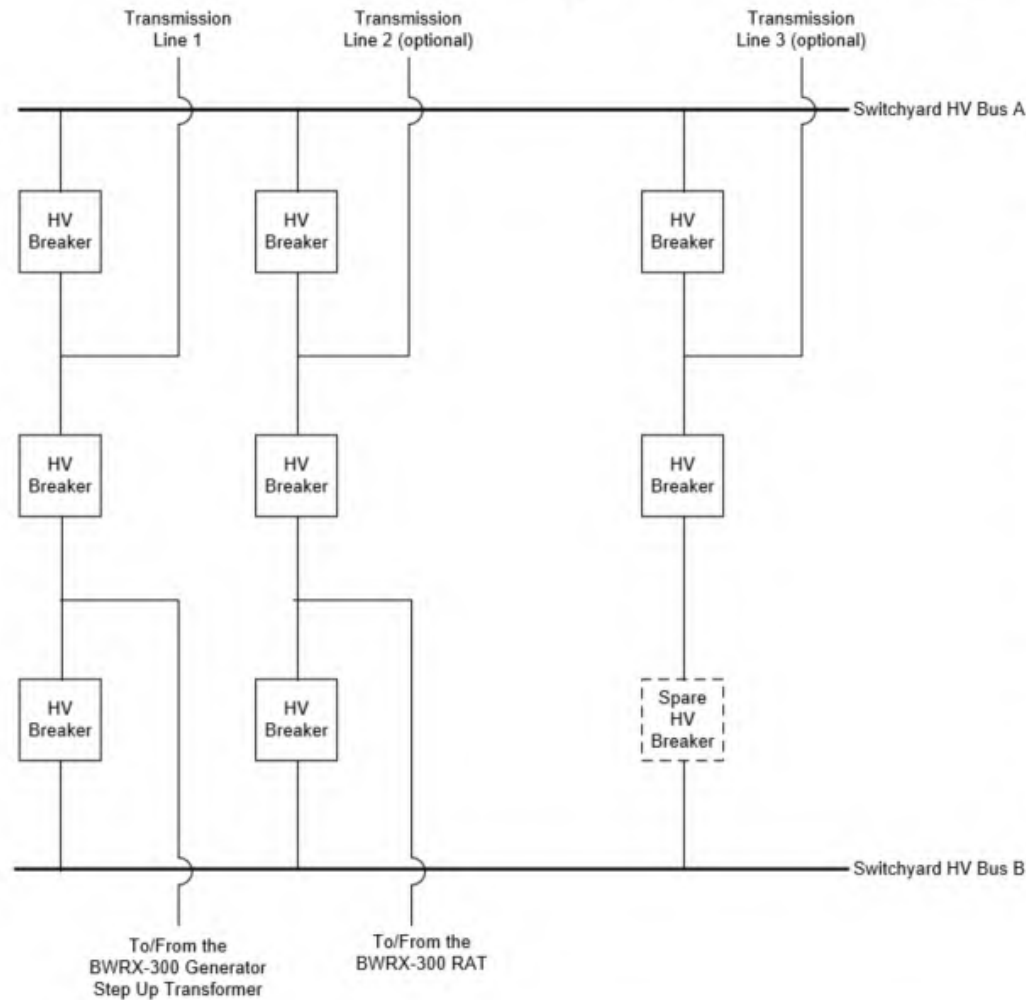
**Figure 8.8-5: Safety Class 2 Low Voltage Uninterruptible Power Supply Distribution**

## SCN MV Busses (A1 and A2) and (B1 and B2)



**Figure 8.8-6: Non-Safety Class Medium Voltage Busses A1-A2 and B1-B2**

### Typical BWRX-300 Switchyard Interface



BWRX-300 only requires one incoming/outgoing transmission line that must be capable of handling the MVA plant output.

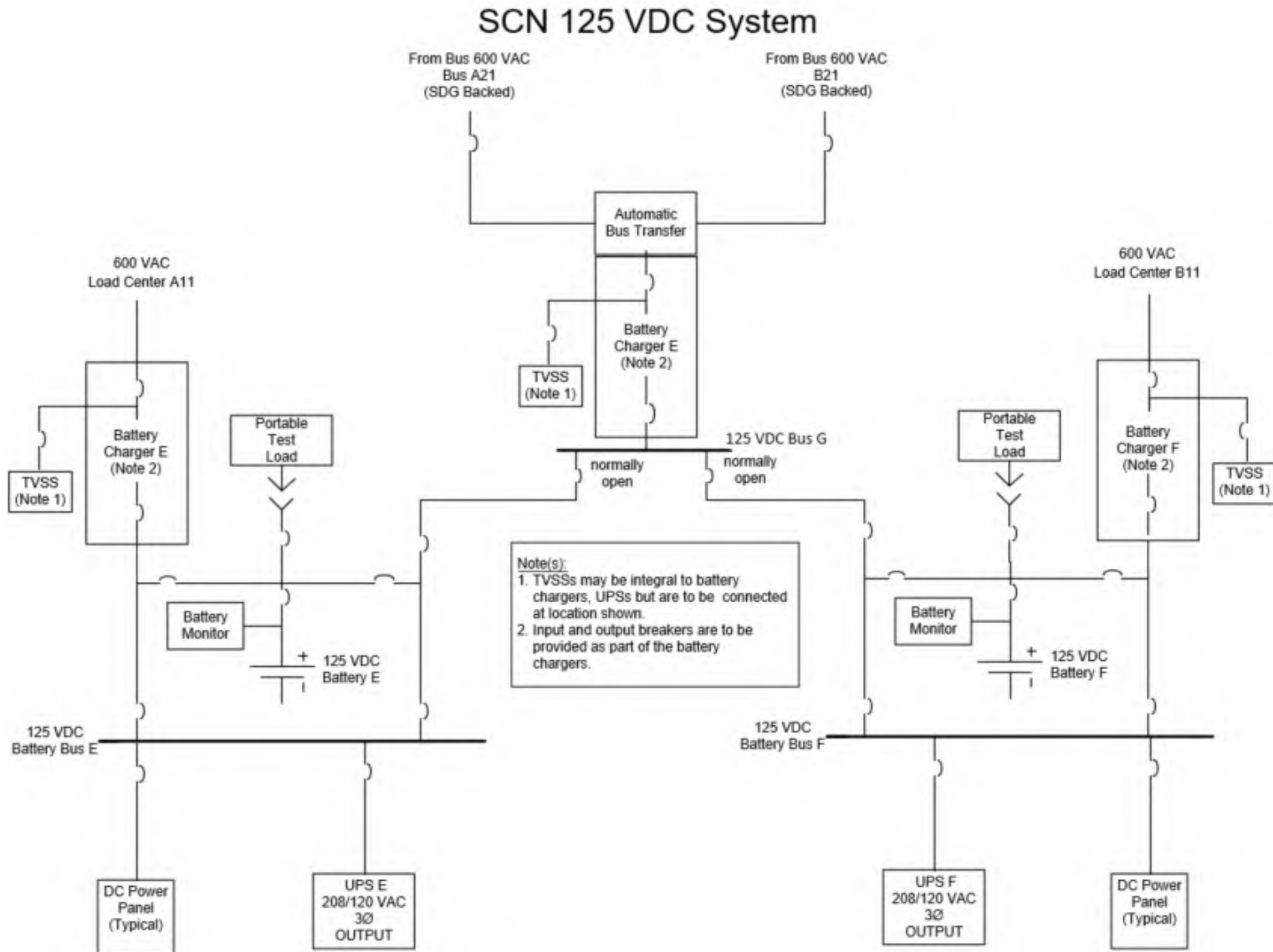
The switchyard may be any voltage or frequency compatible with the plant GSU transformer and may have any number of transmission lines at any voltages at the utility's discretion; the BWRX-300 GSU transformer primary rated voltage will be appropriately changed as needed.

A breaker-and-a-half switchyard breaker scheme as shown is assumed but not required.

Relay house battery chargers and support power should be redundant and can come from a local switchyard transformer or in-switchyard diesel generator, and if necessary, may be powered from a BWRX-300 onsite electrical power system.

Figure 8.8-7: Typical BWRX-300 Switchyard Interface





**Figure 8.8-8: Non-Safety Class 125 VDC System**

## SCN 208/120 VAC System

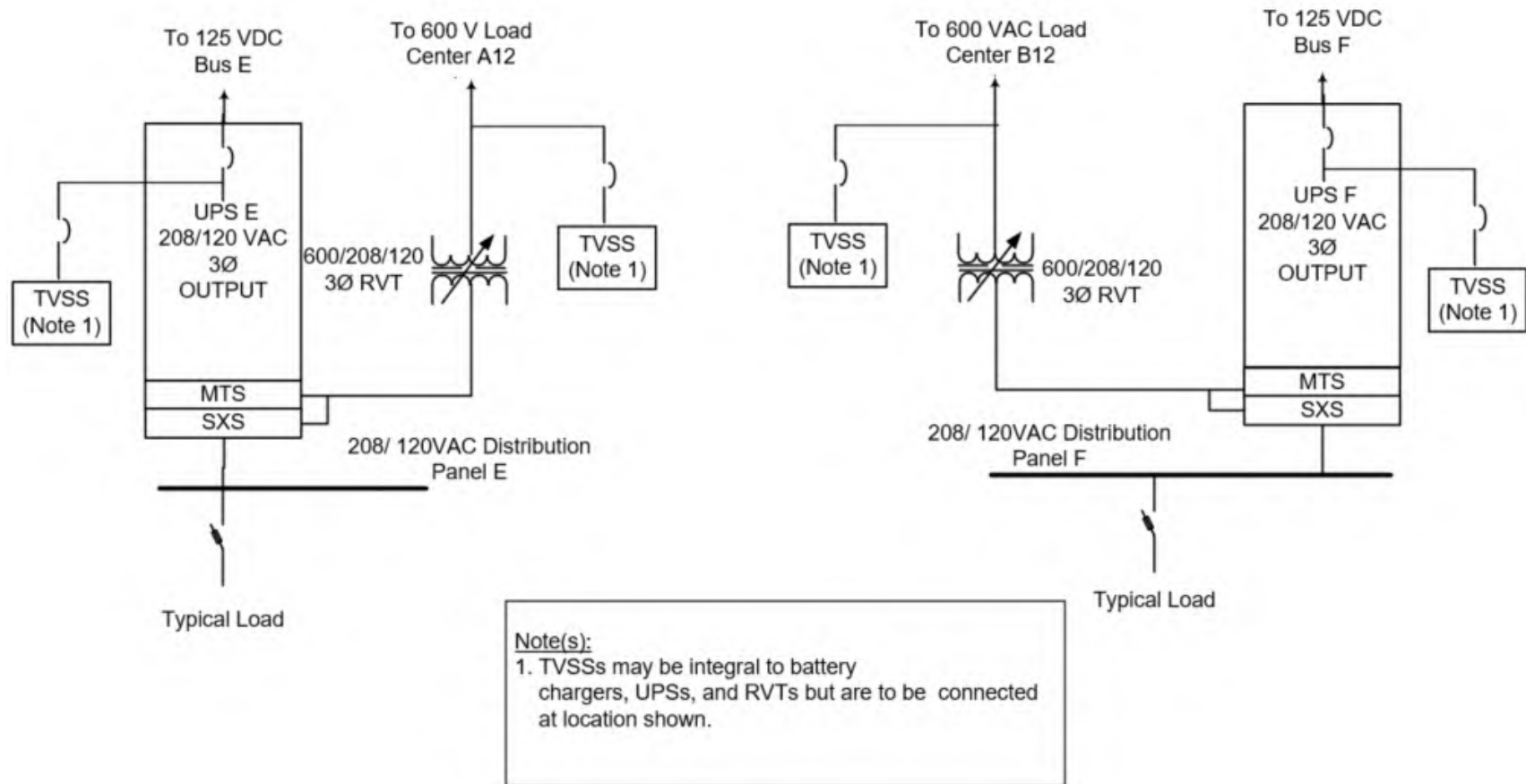


Figure 8.8-9: Non-Safety Class 208/120 VAC System



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33958

Revision 1

October 7, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 9A  
Auxiliary Systems**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	9A.6	Adjustments incorporated per customer acceptance review

### ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
AMCA	Air Movement and Control Association
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ARI	Air Conditioning and Refrigeration Institute
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
BIS	Boron Injection System
BOP	Balance of Plant
BPVC	Boiler and Pressure Vessel Code
CB	Control Building
CCS	Containment Cooling System
CNSC	Canadian Nuclear Safety Commission
CFD	Condensate Filters and Demineralizers System
CFS	Condensate and Feedwater Heating System
CHE	Cranes, Hoists, and Elevators
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CON	Primary Containment System
CRD	Control Rod Drive
CRE	Control Room Envelope
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
CWS	Circulating Water System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCIS	Distributed Control and Information System

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
DEC	Design Extension Condition
DL	Defense Line
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a
DL4b	Defense Line 4b
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
EHC	Electro-Hydraulic Control
EME	Emergency Mitigating Equipment
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
FCU	Fan Coil Unit
FE	Flow Element
FHA	Fuel Handling Accident
FHA	Fire Hazards Assessment
FPC	Fuel Pool Cooling and Cleanup System
FPP	Fire Protection Program
FPS	Fire Protection System
FT	Flow Transmitter
FW	Feedwater
HCW	High Conductivity Waste
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating Ventilation and Cooling System
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	ICS Pool Cooling and Cleanup System

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
ISI	In-Service Inspection
IST	In-Service Testing
LED	Light Emitting Diode
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MSR	Moisture Separator Reheater System
MSRIV	Main Steam Reactor Isolation Valves
MTE	Main Turbine Equipment
NBCC	National Building Code of Canada
NBS	Nuclear Boiler System
NFPA	National Fire Protection Association
NHS	Normal Heat Sink
NPSH	Net Positive Suction Head
OGS	Offgas System
OLNC	On-Line NobleChem™
PAM	Post-Accident Monitoring
PAS	Plant Automation System
PCW	Plant Cooling Water System
PLSA	Plant Services Area
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
RB	Reactor Building
RBS	Reactor Building Structure
RCPB	Reactor Coolant Pressure Boundary
RES	Refueling and Servicing Equipment System



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
RFP	Reactor Feed Pump
RLC	Reactor Level Control
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejector
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SWM	Solid Waste Management System
TB	Turbine Building
TBS	Turbine Building Structure
TMR	Triple Modular Redundant
TS	Technical Specifications
UPS	Uninterruptible Power Supply
WGC	Water, Gas and Chemical Pads

## TABLE OF CONTENTS

<b>9A.0 AUXILIARY SYSTEMS .....</b>	<b>9A-1</b>
9A.1 Fuel Storage and Handling Systems.....	9A-1
9A.1.1 New Fuel Storage and Handling System .....	9A-1
9A.1.2 Fuel Storage and Handling System .....	9A-1
9A.1.3 Fuel Pool Cooling and Cleanup System .....	9A-15
9A.1.4 Handling Systems for Fuel Cask Loading .....	9A-25
9A.2 Water Systems .....	9A-29
9A.2.1 Plant Cooling Water System .....	9A-29
9A.2.2 Reactor Water Cleanup System .....	9A-40
9A.2.3 Shutdown Cooling System .....	9A-53
9A.2.4 Chilled Water Equipment System .....	9A-70
9A.2.5 Normal Heat Sink System .....	9A-82
9A.2.6 Isolation Condenser System Pool Cooling and Cleanup System .....	9A-85
9A.3 Process Auxiliary Systems .....	9A-99
9A.3.1 Process Sampling Systems .....	9A-99
9A.4 Air and Gas System .....	9A-106
9A.4.1 Plant Pneumatic System .....	9A-106
9A.4.2 Containment Inerting System.....	9A-116
9A.5 Heating, Ventilation and Air Conditioning Systems.....	9A-127
9A.5.1 Reactor Building Heating, Ventilation and Air Conditioning System ..	9A-127
9A.5.2 Control Building Heating, Ventilation and Air Conditioning System ...	9A-140
9A.5.3 Radwaste Building Heating, Ventilation and Air Conditioning Systems .....	9A-151
9A.5.4 Turbine Building, Heating, Ventilation and Air-Condition System .....	9A-159
9A.5.5 Plant Services Area Heating, and Ventilation Systems.....	9A-168
9A.5.6 Containment Cooling System .....	9A-175
9A.6 Fire Protection Systems .....	9A-182
9A.6.1 System and Equipment Functions .....	9A-183
9A.6.2 Safety Design Bases.....	9A-184
9A.6.3 Description .....	9A-193
9A.6.4 Materials .....	9A-193
9A.6.5 Interfaces with Other Equipment or Systems.....	9A-193
9A.6.6 Fire Protection System and Equipment Operation.....	9A-194

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

9A.6.7	Instrumentation and Control.....	9A-204
9A.6.8	Monitoring, Inspection, Testing, and Maintenance .....	9A-205
9A.6.9	Radiological Aspects.....	9A-205
9A.6.10	Performance and Safety Evaluation .....	9A-205
9A.6.11	References.....	9A-216
9A.7	Supporting Systems for Diesel Generators (Storage and Transfer, Cooling Water and Cooling Air, Starting, Lubrications, Combustion Air Intake and Exhaust).....	9A-218
9A.7.1	System and Equipment Functions .....	9A-218
9A.7.2	Safety Design Bases.....	9A-218
9A.7.3	Description .....	9A-219
9A.7.4	Materials .....	9A-220
9A.7.5	Interfaces with Other Equipment or Systems.....	9A-220
9A.7.6	System and Equipment Operation .....	9A-220
9A.7.7	Instrumentation and Control.....	9A-220
9A.7.8	Monitoring, Inspection, Testing, and Maintenance .....	9A-220
9A.7.9	Radiological Aspects.....	9A-220
9A.7.10	Performance and Safety Evaluation .....	9A-220
9A.7.11	Diesel Engine Emissions .....	9A-220
9A.7.12	References.....	9A-220
9A.8	Overhead Lifting Equipment.....	9A-221
9A.8.1	Cranes, Hoists and Elevators .....	9A-221
9A.8.2	Fuel Building Crane .....	9A-231
9A.9	Miscellaneous Auxiliary Systems .....	9A-232
9A.9.1	Communication Systems .....	9A-232
9A.9.2	Lighting and Servicing Power System .....	9A-236
9A.9.3	Floor Drain System .....	9A-239
9A.9.4	Interfacing Water Systems.....	9A-248
9A.9.5	Makeup Water System.....	9A-250
9A.9.6	Sanitary Water Systems .....	9A-254
9A.9.7	Chemistry Systems .....	9A-256
9A.9.8	Other Process Media and Other Materials.....	9A-256
9A.9.9	NobleChem™ Injection .....	9A-258
9A.9.10	Chemical Bases of Water Treatment .....	9A-259

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

9A.10Storage System for Non-Permanent Equipment Used in Design Extension Conditions .....	9A-260
--	--------

## LIST OF TABLES

Table 9A.1.3-1: Fuel Pool Cooling and Cleanup System Interfaces .....	9A-22
Table 9A.2.1-1: Plant Cooling Water System Interfaces.....	9A-37
Table 9A.2.2-1: Reactor Water Cleanup System Interfaces .....	9A-49
Table 9A.2.2-2: Reactor Water Cleanup System Operating Modes .....	9A-51
Table 9A.2.3-1: Shutdown Cooling System Interfaces .....	9A-63
Table 9A.2.3-2: BWRX-300 Shutdown Cooling System Mode Table .....	9A-65
Table 9A.2.3-3: Shutdown Cooling Configuration During Refueling Mode .....	9A-66
Table 9A.2.4-1: Chilled Water Equipment System Interfaces.....	9A-79
Table 9A.2.5-1: Normal Heat Sink Interfaces .....	9A-84
Table 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System Interfaces .....	9A-93
Table 9A.2.6-2: Isolation Condenser System Pool Cooling and Cleanup System Operating Modes .....	9A-95
Table 9A.2.6-3: Isolation Condenser System Pool Cooling and Cleanup System Instrumentation .....	9A-96
Table 9A.3.1-1: Process Sampling Subsystem Interfaces.....	9A-104
Table 9A.4.1-1: Plant Pneumatics System Interfaces.....	9A-112
Table 9A.4.2-1: BWRX-300 Operational Modes .....	9A-123
Table 9A.4.2-2: Containment Inerting System Interfaces .....	9A-124
Table 9A.5.1-1: Reactor Building Heating, Ventilation and Air Conditioning System Interfaces .....	9A-137
Table 9A.5.2-1: Control Building Heating, Ventilation and Air Conditioning System Interfaces.....	149
Table 9A.5.3-1: Radwaste Building Heating, Ventilation and Air Conditioning System Interfaces .....	9A-157
Table 9A.5.4-1: Turbine Building, Heating, Ventilation and Air-Condition System Interfaces .....	9A-166
Table 9A.5.5-1: Plant Services Area Heating, and Ventilation System Interfaces .....	9A-173
Table 9A.5.6-1: Containment Cooling System Interfaces .....	9A-180
Table 9A.6.2-1: Applicable BWRX-300 Fire Protection Design Codes and Standards and Key Documents .....	9A-186
Table 9A.6.2-2: System Interfaces.....	9A-191
Table 9A.6.2-3: FPS Component Design Characteristics.....	9A-217
Table 9A.8.1-1: Cranes, Hoists, and Elevators Components .....	9A-229
Table 9A.8.1-2: Cranes, Hoists and Elevators System Interfaces .....	9A-230

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Table 9A.9.3-1: Equipment and Floor Drain System Interfaces.....	9A-244
Table 9A.9.5-1: Makeup Water System Interfaces .....	9A-253

## LIST OF FIGURES

Figure 9A.1.2-1: Fuel Pool Arrangement .....	9A-14
Figure 9A.1.3-1: Fuel Pool Cooling and Cleanup System .....	9A-24
Figure 9A.2.1-1: Plant Cooling Water .....	9A-39
Figure 9A.2.2-1: Reactor Water Cleanup System.....	9A-52
Figure 9A.2.3-1: Decay Heat Removal Subsystem .....	9A-67
Figure 9A.2.3-2: Overboard Subsystem .....	9A-68
Figure 9A.2.3-3: Thermal Stratification Reduction Subsystem .....	9A-69
Figure 9A.2.4-1: Chilled Water Equipment .....	9A-81
Figure 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System .....	9A-98
Figure 9A.4.1-1: Plant Pneumatic System .....	9A-115
Figure 9A.4.2-1: Containment Inerting System.....	9A-126
Figure 9A.5.1-1: Reactor Building HVAC.....	9A-139
Figure 9A.5.2-1: Control Building HVAC.....	9A-150
Figure 9A.5.3-1: Radwaste Building HVAC .....	9A-158
Figure 9A.5.4-1: Turbine Building HVAC .....	9A-167
Figure 9A.5.5-1: Plant Services Area HVAC.....	9A-174
Figure 9A.5.6-1: Containment Cooling System.....	9A-181
Figure 9A.6.6-1: BWRX-300 Fire Protection System Schematic .....	9A-195
Figure 9A.6.10-1: BWRX-300 Fire Hazard Assessment Methodology .....	9A-212
Figure 9A.6.10-2: BWRX-300 Fire Safe Shutdown Analysis Methodology.....	9A-214
Figure 9A.9.3-1a: Floor Drain System Drain Sump (Typical).....	9A-246
Figure 9A.9.3-1b: Floor Drain System Containment Drain Sump (Pressurized).....	9A-247

## **9A.0 AUXILIARY SYSTEMS**

### **9A.1 Fuel Storage and Handling Systems**

#### **9A.1.1 New Fuel Storage and Handling System**

New fuel receiving is a service and a material handling process that utilizes the Fuel Handling System and equipment described in Subsections 9A.1.2, 9A.1.4 and 9A.8. New fuel receipt is described in Section 5.1 of the Safeguards Annex. The BWRX-300 design does not employ a new fuel storage vault. The new fuel assemblies are stored in fuel storage racks which are located in the Fuel Pool in the Reactor Building. New fuel and spent fuel are stored in the same Fuel Pool. The following information is provided in support of the requirements of CNSC REGDOC-2.5.2 Section 8.12.1 (Reference 9A.1.1-1) relative to new fuel storage and handling:

1. Nuclear criticality safety, refer to Subsection 9A.1.2.3.1
2. Maintenance, periodic inspection, and testing of components important to safety, refer to Subsection 9A.1.2.8 and Section 5 of the Safeguards Annex
3. Inspection of non-irradiated fuel, refer to the Safeguards Annex, Section 5.1.2 and Chapter 4, Subsection 4.2.5.
4. Prevent loss of or damage to the fuel, refer to Subsection 9A.1.2.3.13 and Safeguards Annex Section 7 for information pertaining to prevention of loss of fuel. Subsection 9A.1.4 addresses fuel handling equipment and Subsection 9A.8 addresses movement of heavy loads in the vicinity of the fuel pool
5. Meet Canada's safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to non-irradiated fuel containing fissile material, refer to Safeguards Annex, Section 4 "Safeguards and Operation" and Section 7 "Nuclear Material Accountancy"

##### **9A.1.1.1 References**

9A.1.1-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants"

#### **9A.1.2 Fuel Storage and Handling System**

The fuel pool is used to store spent fuel after discharging and new fuel after delivery to the site and before core loading. The above grade portion of the Reactor Building (RB) structure houses the refueling floor, refueling and fuel handling systems, fuel pool, and Reactor Building Polar Crane.

##### **9A.1.2.1 System and Equipment Functions**

System and equipment functions associated with the Fuel Storage and Handling System include the followings:

##### **Normal Functions (Non-Safety-Category)**

- Permit disassembly and re-assembly of containment and reactor.
- Permit reactor refueling.
- Permit new and spent fuel storage in fuel pool.
- Permit replacement of reactor mechanical components.



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- Permit replacement of nuclear instrumentation.
- Support various outage maintenance.
- Maintain subcriticality during fuel and core component handling operations.

**Normal Functions (Safety-Category)**

- Fuel Storage and Handling System does not perform any Safety Category functions during off-normal conditions.

#### **Off-Normal Functions (Non-Safety-Category)**

- Fuel Storage and Handling System does not perform any Non-Safety-Category functions during off-normal conditions.

#### **Off-Normal Functions (Safety-Category)**

- Maintain Fuel Pool structural integrity following a seismic event
- Fuel Storage and Handling System does not perform any Safety-Category functions during off-normal conditions.

#### **9A.1.2.2 Safety Design Bases**

The fuel pool is adequately protected against dynamic effects. The fuel pool is designed to ensure that required water levels are maintained in normal and off-normal conditions. Fuel in the fuel pool is protected from damage caused by the postulated drop of fuel assemblies, bundles, or other objects onto stored fuel (Subsections 9A.1.4 and 9A.8) by assessing the impact of the load drop and designing systems to mitigate the effects.

The orientation and location of the turbine shafts are designed such that the fuel storage components are located outside the turbine missile low-trajectory strike zone (Chapter 10, Subsection 10.2.3). Furthermore, the turbine is located in the Turbine Building with Turbine Building and Reactor Building walls between the Fuel Pool and Turbine.

The fuel pool has the following design features: control for airborne release of radioactive material; drains, gates, and weirs that prevents drainage of coolant inventory below an adequate shielding depth; adequate coolant flow provided to the fuel racks; and a system to detect fuel pool leakage.

The addition of water to the fuel pool does not result in a criticality event.

A full array in the loaded fuel rack is designed to be subcritical by at least 5%  $\Delta k$  under all normal and abnormal conditions as per CNSC REGDOC-2.4.3 Section 2.3 (Reference 9A.1.2-1). Safety analyses are performed using validated codes (Chapter 3, Appendix 3E).

Fuel pool storage design meets Canadian Nuclear Safety Commission (CNSC) requirements specified in CNSC REGDOC-2.5.2 Section 8.12 (Reference 9A.1.2-2) as related to ensuring the design and handling of irradiated fuel.

#### **9A.1.2.3 Description of System**

The BWRX-300 Steel-plate Composite Containment Vessel (SCCV) and the main RB structure including the reactor cavity, equipment pool and fuel pool are classified as Safety Class 1 (SC1).

The fuel pool base mat is located at approximately the power block grade elevation. The fuel pool extends up to the refuel floor elevation of approximately 13 metres above grade elevation.

The fuel pool size is determined by the volume of cooling water, size of the fuel storage racks, with control rod blade guides, other equipment required to be stored in the pool, and lay-down areas. Refueling outage equipment such as lights, test weights, dummy fuel bundles, and control blades are also stored in the fuel pool. The fuel preparation machine is located on the fuel pool periphery. The new fuel inspection stand is mounted on the refuel floor away from the fuel pool periphery and only used for new fuel before it is installed into the water and to the fuel prep machine. Used reactor core instruments may also be stored in the fuel pool, until such time as they are cut up and packaged.

Additional information pertaining to the design of the new and spent fuel storage system is presented below.

#### **9A.1.2.3.1 Nuclear Criticality Safety**

Chapter 15, Subsection 15.5.9.3 analyzes the dose consequence of a representative Out-of-Core-Criticality accident scenario compliant with the requirements of CNSC REGDOC-2.4.3 (Reference 9A.1.2-1).

New and spent fuel are stored in storage racks capable of maintaining fuel subcritical. The fuel pool is comprised of a deep pit filled with water which provides shielding in addition to the shielding provided by the pool confinement structure. Within the pool are storage racks in a grid pattern which contain a fixed neutron absorbing material. The storage rack geometry and material are designed to preclude the possibility of criticality under normal and credible abnormal conditions.

A full array in the loaded fuel rack is designed to be subcritical by at least 5%  $\Delta k$ . Monte Carlo techniques are employed in the calculations performed to assure that  $k_{\text{eff}}$  does not exceed 0.95 under all normal and abnormal conditions.

#### **9A.1.2.3.2 Heat Removal in Operational States, Design Basis Accidents and Design Extension Conditions**

Removal of heat from the fuel pool during operational and design basis accident conditions is described in Subsection 9A.1.3. The design of the Fuel Pool contains provisions for Fuel Pool Design Extension Conditions (DECs) by:

1. Ensuring that boiling in the pool does not result in structural damage (Chapter 3, Subsection 3.5.6.1)
2. Providing temporary connections to enable the refill of the pool using temporary supplies (Subsection 9A.1.3.6.4)
3. Providing temporary connections to heat removal systems for power and cooling water (Subsection 9A.1.3.3 discusses redundant cooling trains)
4. Ensuring that the design of the fuel pool is such that a Fuel Handling Accident does not exceed site Design Basis Accident dose acceptance criteria Chapter 15, Subsection 15.5.8
5. Ensuring that severe accident management actions related to the fuel pool can be carried out (Chapter 15, Subsection 15.1.5)

#### **9A.1.2.3.3 Inspection of Irradiated Fuel**

The handling and storage systems are used primarily to support refueling and core shuffling, and the same equipment supports periodic inspections as needed of irradiated fuel. The typical inspection scenario entails the Refueling Platform (Subsection 9A.1.2.3.5) moving fuel to the Fuel Preparation Machine. The Fuel Preparation Machine is typically outfitted with a fuel inspection fixture which is a small rotator installed onto the Fuel Preparation Machine to facilitate fuel inspections. In addition to supporting the loading of new fuel into the fuel pool during outage preparation, the Fuel Preparation Machine supports activities associated with the inspection of irradiated fuel, including removal and reinstallation of the fuel channel to enable access to the bundle/rods. Typically located adjacent to the Fuel Preparation Machine there is a channel handling boom, and below the surface of the fuel pool on the wall, a channel storage rack. When irradiated fuel is moved into the Fuel Preparation Machine for inspection, procedural requirements specify the water coverage that must be maintained over the top of the irradiated fuel. Typically, a mechanical interlock on the drive of the Fuel Preparation Machine is utilized to limit or block raising the bundle above an approved radiologically safe elevation. Plant equipment utilized for servicing irradiated fuel includes the following.

### **Fuel Preparation Machine**

Two Fuel Preparation Machines are provided for handling fuel assemblies while removing or installing channels on the fuel bundles. Each machine consists of a fuel bundle carriage, which rides on a frame that is mounted on the edge of the fuel pool and extends down to depth along the pool wall and a work platform. The work platform includes handrails and extends out over the pool to facilitate viewing of the channeling operations.

### **Fuel Inspection Fixture**

The Fuel Inspection Fixture is used in conjunction with the Fuel Preparation Machine to permit remote inspection of fuel elements. The Fuel Inspection Fixture, when installed, permits rotation of the fuel assembly in the carriage, and in conjunction with the vertical movement of the carriage provides complete access to all quadrants of fuel assembly inspection.

### **New Fuel Inspection Stand**

The New Fuel Inspection Stand is stored on the refueling floor of the reactor building. It consists of a single "U" shaped platform with guardrails, a column which supports two fuel assemblies in a vertical position, and a personnel lift which raises or lowers the work platform along the fuel assemblies for inspection purposes. The lower base of the fuel assembly sits in the fuel seat at the bottom of the stand. The fuel assemblies are mounted on rotatable bearing surfaces which allow for rotation to desired quadrants for inspection.

### **Channel Handling Tool**

The Channel Handling Tool is used in conjunction with the Fuel Preparation Machine to remove, install, and handle fuel channels in the fuel storage pool. The bail hangs from a load balancer on the channel handling boom mounted adjacent to the Fuel Prep Machines.

### **Channel Transfer Grapple**

The Channel Transfer Grapple is an air actuated device consisting of a frame, air cylinder, and two jaws. It is used with the Refueling Platform auxiliary hoist to transport individual irradiated fuel channels between working and storage facilities in the fuel pool.

### **Channel Bolt Wrench**

The Channel Bolt Wrench is a manually operated device used for removing and installing the Channel Fastener.

### **Channel Handling Boom**

The Channel Handling Boom is located in a socket adjacent to the fuel preparation machines and adjacent to the fuel pool. The boom supports the channel handling tool and spring balancer over the Fuel Preparation Machine during channel removal and installation operations.

### **Inspection Camera**

Cameras and tooling for inspections and measurements are curb-mounted next to the Fuel Preparation Machine in support of underwater inspection activities. Assorted hand tools (handling poles) and grapples are used in inspections. Sometimes, inspections are conducted during outages, other times, inspections on discharged fuel occurs post-outage (during the operating cycle).

#### ***9A.1.2.3.4 Periodic Inspection and Testing of Components Important to Safety***

For information pertaining to periodic inspection and testing of components refer to Subsection 9A.1.2.8.

#### **9A.1.2.3.5 Design for Precluding the Dropping of Irradiated Fuel in Transit**

The Reactor Building is supplied with a Refueling Platform for fuel movement and reactor servicing support tasks. Fuel and other components are removed from the reactor core, transported to the fuel pool, and then returned to the reactor as required. Additionally, the Refueling Platform serves as a service area which is used to assist in normal vessel disassembly/reassembly, normal Reactor Pressure Vessel maintenance activities, and in performance of long-term in-vessel inspection and maintenance activities.

The Refueling Platform consists of a bridge which spans the width of the Fuel Pool and Reactor Cavity. A trolley rides on the bridge, traversing the width of the bridge to the maximum width. A full length sized working platform where refueling, servicing, and inspection personnel perform their work support tasks spans the length of the bridge. The Refueling Platform rides on four wheels running on two rails set into the refuel floor.

The Refueling Platform is equipped with a traversing trolley, an operator control console, a main hoist with a telescoping mast and fuel grapple, an auxiliary, and a monorail hoist. An air compressor on the Refueling Platform provides compressed air to the pneumatic system for the fuel grapple and reactor service tooling. A frame mounted hoist and a monorail hoist support handling the smaller core components and tooling.

The Refueling Platform is a rigid structure built to ensure accurate and repeatable positioning during the refueling process. The telescoping mast and grapple are suspended from a trolley system and is used to lift and orient fuel bundles for placement in the core or storage racks. The Refueling Platform includes a control console providing the operator precise control of the motions of bridge, trolley, and all hoists. Bridge, trolley, and mast grapple elevation readouts are located in a clearly understandable and convenient position for the operators to monitor as they position the Refueling platform. The control console includes a light indication to monitor hoist functions and refueling interlocks.

The main hoist and fuel grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Specialized grapple tooling which carries fuel assemblies or control rod blades is designed such that accidental opening is mechanically prevented and, upon loss of power, the grapples fail in the closed position.

Interlocks on the Refueling Platform prohibit bridge movement towards the core when the reactor mode switch is in refueling position and a single control rod is out.

Power shuts off to the main fuel hoist motor when the Refueling Platform is over the core, the hoist load cell is sensing load of a fuel assembly or greater, and the reactor mode switch is in refueling and a single rod is withdrawn.

Hoist motion is prevented when the grapple hooks are not closed, and the fuel hoist senses load of a fuel assembly or greater.

Upon completion of fuel movement operations, a core verification task is accomplished in which individual fuel assemblies are verified to be installed in the correct position using serial numbers assigned to the fuel assemblies. Refer to Subsection 9A.1.4 for information pertaining to the design of equipment and operations associated with precluding dropping of irradiated fuel while in transit.

Spent fuel is transferred from the fuel pool in a Transfer Cask using the single failure proof RB Polar Crane. Refer to Subsection 9A.8.1 for information pertaining to the design and operation of the RB Polar Crane as it relates to precluding the dropping of irradiated fuel while the in transit.

#### **9A.1.2.3.6 Handling Stresses on Fuel Elements or Fuel Assemblies**

The structural adequacy of the fuel assembly components is demonstrated by evaluations (analysis or testing) that specifically address the operational duty that results from the BWRX-300 environment. This duty results from steady-state operation (including handling loads), mechanical loads associated with anticipated transients, and accident loads due to external conditions.

The fuel assembly structural components are evaluated to ensure that the components do not fail due to stresses exceeding the fuel assembly component mechanical capability.

##### **Upper Tie Plate**

The design loading for the Upper Tie Plate is from bundle handling. Specifically, a load equal to three times the bundle weight is applied at the tie plate handle to grapple attachment. The load is reacted at the eight tie-rod locations.

##### **Lower Tie Plate**

The design loading for the Lower Tie Plate is from bundle handling.

##### **Fuel Rod End Plugs**

The design loading for the Fuel Rod End Plugs is from bundle handling.

##### **Plenum Spring**

The Plenum Spring is designed to resist an acceleration of the fuel pellet column while being transported without deflecting the spring.

##### **Expansion Spring**

The Expansion Springs are designed to resist downward forces from grappling and the weight of the suspended components such as the tie plate while allowing expansion from irradiation growth of the individual fuel rods.

##### **Water Rods**

The Water Rod tubing was evaluated for a steady-state differential wall pressure. The maximum load that a Water Rod tab could experience due to operating effects of spacer lift forces from flow or differential thermal expansion between the fuel rods and Water Rods is the load required to simultaneously slide all fuel rods through a spacer.

##### **Spacer**

Tests are performed to demonstrate that the Spacer design can withstand design basis loading without any significant deformation.

##### **Channel**

The design loadings for the Channel include steady-state and transient operating differential pressure. The Channel is tested to demonstrate capability to withstand bending loads from lateral seismic loading.

#### **9A.1.2.3.7 Inadvertent Dropping of Heavy Objects and Equipment on Fuel Assemblies**

Refer to Subsection 9A.8.1 and Chapter 3, Subsection 3.4.4.3, for information pertaining to the inadvertent dropping of heavy objects and equipment on fuel assemblies.

#### **9A.1.2.3.8 Inspection and Safe Storage of Suspect or Damaged Fuel Elements or Fuel Assemblies**

Although rare, mechanical damage to a fuel assembly can occur due to fuel rod cladding wear from fretting or from a fuel handling accident. In the unlikely event that a Fuel Handling Accident (FHA) does occur, visual inspections are performed to assess the extent of damage followed by an action plan. The damaged assembly is recovered under a special written procedure lifting from the normal bail handle and stored in a fuel storage rack.

Failed fuel requires special consideration because of its significance during operations. Fuel failures are first detected via the plant's Offgas System (Chapter 11, Subsection 11.3). In addition, grab samples for measuring individual isotopic activities with spectrometry equipment is typically performed.

Typically, six major nuclides are measured and the sum of these is reported as the total offgas release rate. Xenon-133 is by far the most important isotope among these for determining the presence of fuel failures. Virtually all "indicators" or calculated parameters for fuel integrity monitoring rely on the relative increase in Xe-133 inventory to help detect leaking spent fuel. Whenever the measured isotopic mixture becomes relatively rich in Xe-133 it indicates a cladding perforation and release of stored gas. Fission product activity in the reactor coolant is monitored separately (e.g., iodine, strontium, and cesium isotopes) and typically only shows an increase for more significant failures.

#### **9A.1.2.3.9 Radiation Protection**

Radiation protection complies with the CNSC guidance (Reference 9A.1.2-3) for normal operation and anticipated operational occurrences. Radiation exposure is kept within regulatory limits and As Low As Reasonably Achievable (ALARA) (Chapter 12, Subsection 12.3.1) goals.

Radiation monitoring is provided for the fuel pool storage area and the associated ventilation paths (Chapter 11, Subsection 11.5.3). The Refueling Platform, hoist, and grapple include interlocks to prevent potential refueling errors and reduce the possibility of exposure of plant workers. The Refueling Platform telescoping mast design provides a rigid mechanical stop that makes it physically impossible to raise fuel above the safe level below the water.

The fuel pool is designed to ensure the area dose rate is less than 25  $\mu\text{Sv/hr}$ . Fuel handling equipment design ensures that the operator is not exposed to 25  $\mu\text{Sv/hr}$  when fuel and irradiated components are being stored or handled in accordance with ANSI 57.1, Design Requirements for Light Water Reactor Fuel Handling System (Reference 9A.1.2-5). The BWRX-300 design does not have integrated containment shielding in the fuel transfer zone from the core to fuel pool and over the top of containment. This creates the need for a temporary and removable radiation shielding transfer canal or cattle chute that shields personnel in containment from fuel or irradiated components passing through the fuel transfer zone to the fuel pool. The water coverage over spent fuel assemblies in transit to the fuel pool or core is influenced by the grapple normal up limit and the fuel pool normal water level.

The air above the fuel pool and refuel floor equipment areas is monitored. In the event of a Fuel Handling Accident in the Fuel Handling Machine area, a radiation monitor alarm initiates closure of the RB isolation dampers and securing of the RB upper level supply Air Handling Units.

#### **9A.1.2.3.10 Identification of Individual Fuel Bundles**

Every fuel bundle contains a serial number on top of the bail handle which is part of the upper tie plate. Not only are these serial numbers used to specify each bundle's location in the core for each cycle but ultimately to track every bundle from manufacturing to ultimate disposal. All special

nuclear material accountability rules are strictly followed, and records are retained for every movement of a bundle over the bundle's life. Refer to Safeguards Annex, Section 7 for information pertaining to nuclear material accountancy.

**9A.1.2.3.11 Maintenance and Decommissioning of Fuel Storage and Handling Facilities**

Proper maintenance and operation of Fuel Pool systems is necessary to maintain water quality and radionuclides at acceptable levels. Maintenance of water quality is necessary to prevent degradation of the spent fuel and other materials stored in the fuel pool (i.e., control rod blades or incore instrument strings). Fuel pool water treatment and system maintenance programs prevent the buildup of excessive concentrations of contaminants and radionuclides and mitigate the consequences of any potential release from the fuel pool (Subsection 9A.1.3).

Scheduled maintenance ensures that structures and systems required for containing, cooling, cleaning, level monitoring and makeup of water in the fuel pool are operable consistent with the licensing basis. The application of scheduled maintenance reduces high levels of contaminants and radionuclides in the pool water that can have adverse effects on stored fuel, the fuel pool, fuel transfer components, and related equipment.

Operating procedures coupled with surveillances and observations, indicate changes in fuel pool level. Procedures address appropriate maintenance, calibration, and surveillance of available monitoring equipment.

Plant decommissioning and dismantling activities considered at the design phase include considerations of experience gained from the decommissioning of existing plants, as well as those plants that are in long-term safe storage. Refer to Chapter 21 for information related to decommissioning of the BWRX-300.

**9A.1.2.3.12 Decontamination of Fuel Handling and Storage Areas and Equipment**

Decontamination of fuel handling and storage areas and equipment is performed as required to comply with ALARA requirements as discussed in Chapter 12 and prevent the spread of contamination.

New and spent fuel is stored in the fuel pool. As part of typical operating practice, area's outside of the fuel pool where fuel is handled are surveyed for contamination. The plant is supplied with equipment and features to accomplish effective decontamination without spreading contamination. Wash-down areas and sink drains are routed to the liquid radioactive waste system (Chapter 11, Subsection 11.2). The Reactor Building Ventilation System (Subsection 9A.5.1) maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere. Vendor-supplied services may also be utilized for handling and storage area decontamination.

In the event radiation surveys of tools indicate additional decontamination is required the tools may be sent to a decontamination room for processing.

Design considerations for environment and safety are imposed upon tooling to facilitate operator safety and to ensure ALARA guidance and best practices are incorporated into the design process. Typical design considerations for the decontamination of fuel handling and storage areas and equipment includes:

1. Ease of decontamination
2. Elimination of crevices which facilitates crud removal



3. Minimization of crud buildup which facilitates decontamination and cleaning of equipment, components, and contaminated areas, and helps to prevent airborne contamination dispersion
4. Use of the best construction materials (smooth surface)
5. Use of HEPA filters as required to support cleaning of equipment post use

Criteria for selecting tools, materials, and equipment for use in contaminated areas includes minimizing the use of porous or other materials that are difficult to decontaminate.

**9A.1.2.3.13 Implementation of Adequate Operating and Accounting Procedures to Prevent Loss of Fuel**

Refer to Safeguards Annex, Section 7, for information related to operating and accounting procedures to prevent loss of fuel.

CNSC regulatory document REGDOC-2.13.1 (Reference 9A.1.2-4), on safeguards and nuclear material accountancy, sets out CNSC's requirements and guidance for the:

- Establishment and maintenance of a safeguards program
- Requirements for event and compliance monitoring reporting by licencees

OPG compliance with CNSC REGDOC-2.13.1 is discussed in the "Safeguards Annex."

**9A.1.2.3.14 Measures to Prevent a Direct Threat or Sabotage to Irradiated Fuel**

Information pertaining to the measures to prevent a direct threat or sabotage to irradiated fuel is considered Protected Information and withheld from the public.

**9A.1.2.3.15 Safeguards Requirements for Recording and Reporting Accountancy Data, and for Monitoring Flows and Inventories Related to Irradiated Fuel Containing Fissile Material**

Refer to the Safeguards Annex, Section 4 "Safeguards and Operation" and Section 7 "Nuclear Material Accountancy" for information pertaining to safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to irradiated fuel containing fissile material.

The fuel pool which is used for fuel storage includes provisions as discussed in the noted sections:

1. Controlling the chemistry and activity of any water in which irradiated fuel is handled or stored (Subsection 9A.1.3, Table 9A.1.3-1)
2. Monitoring and controlling the water level in the fuel storage pool (Subsection 9A.1.3)
3. Detecting Leakage (Subsection 9A.1.2.3.16)
4. Preventing the pool from emptying in the event of a pipe break (Subsection 9A.1.2.2)
5. Sufficient space to accommodate the entire reactor core inventory at all times (Subsection 9A.1.2.3.16)

**9A.1.2.3.16 Component Description**

**Storage Racks**

The fuel racks consist of a stainless steel structure composed of neutron absorbing material in a series of square vertical tubes (cells).

The fuel storage racks are top entry racks with bail extended above the rack and designed to preclude the possibility of criticality.

The fuel assemblies are stored in the fuel storage racks that are arranged in the fuel pool as shown in Figure 9A.1.2-1. The fuel storage racks provide space for storage of fresh fuel as well as space for longer term storage of the spent fuel assemblies for cooling prior to transfer to onsite storage or off-site shipment.

The BWRX-300 design includes storage for approximately 600 bundles. The fuel racks can hold an entire reload of fresh fuel and up to approximately eight years of spent fuel and have space remaining to accept a full core of off-loaded fuel.

### **Liner**

The pool liner is designed to meet Seismic Category B requirements. The fuel pool liner is capable of withstanding all design loads as discussed in Chapter 9B. Under certain conditions the fuel pool boils to provide cooling of the spent fuel. The design of the fuel pool liner is capable of withstanding the high temperatures associated with a boiling pool. A leak detection system is provided to detect fuel pool leaks.

For normal conditions of operation, the spent fuel assemblies are cooled by the Fuel Pool Cooling and Cleanup System (FPC). In the event of a loss of the fuel pool cooling and cleanup functions (such as station blackout) the primary defense line maintaining fuel pool temperature is boiling of the fuel pool. Refer to Subsection 9A.1.3 for a description of the FPC.

### **Refueling Platform**

The refueling platform is a gantry-type crane that spans the reactor vessel cavity and fuel pool and is employed to handle fuel and perform other ancillary tasks in the reactor building. It is equipped with a traversing trolley on which is mounted a telescoping mast and a fuel grapple. The refueling platform is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refueling process. Additional information pertaining to the refueling platform is provided in Subsection 9A.1.2.3.5.

#### **9A.1.2.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

#### **9A.1.2.5 Interfaces with Other Equipment or Systems**

Interfaces associated with the Fuel Pool include those identified in Subsection 9A.1.3.5 for the Fuel Pool Cooling and Cleanup System as well as the Fuel Handling Systems for Cask Loading, (Subsection 9.1.4), Reactor Building Polar Crane (Section 9A.8), and RB HVAC (Subsection 9A.5.1).

#### **9A.1.2.6 Systems and Equipment Operation**

Refer to Subsection 9A.1.3 for information related to the operation of the Fuel Pool Cooling and Cleanup System. Refer to Subsection 9A.1.4 for information pertaining to fuel handling operations, Subsection 9A.5.1 for information pertaining to Heating, Ventilation and Cooling System, and Section 9A.8 for information pertaining to the operation of cranes, hoists, and elevators.

#### **9A.1.2.7 Instrumentation and Control**

Subsection 9A.1.3 "Fuel Pool Cooling and Cleanup" describes the fuel pool water temperature and water level instrumentation. Chapter 12, Section 12.3 describes design features for radiation protection. Subsection 9A.5.1.8 describes the Heating, Ventilation and Cooling System instrumentation.

#### **9A.1.2.8 Monitoring, Inspection, Testing, and Maintenance**

The design of the fuel pool liner and fuel storage racks facilitate inspections of the exposed surface of the liner and fuel storage rack.

Neutron absorbing material coupons are provided with the spent fuel racks for the purpose of monitoring for potential degradation of the material over the design life of the equipment.

Refer to Chapter 3, Subsection 3.5.5.6 for information pertaining to testing and in-service inspection requirements related to the RB. The design of the fuel pool provides monitoring for the loss of decay heat removal capability using the temperature measuring instruments in the Fuel Pool Cooling and Cleanup System as described in Subsection 9A.1.3. Radiation monitors are provided in the Fuel Pool area to detect both general area radiation levels and airborne contamination levels as described in Chapter 12, Subsection 12.3.5.4.

#### **9A.1.2.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

The design of the fuel pool minimizes buildup of contamination and provides shielding. The surface finishes of the components for the fuel storage racks are smooth to minimize accumulation of radioactive materials and to facilitate surface decontamination. The depth of the water above the fuel assemblies and the thick concrete walls of the fuel pool provides shielding for the assemblies.

#### **9A.1.2.10 Performance and Safety Evaluation**

The fuel storage racks, and fuel pool (including liner) are located inside of the Reactor Building. The design of the Reactor Building withstands combinations of mechanical, hydraulic, and thermal loads and natural phenomena effects, including severe winds such as hurricanes and tornadoes (Chapter 3, Subsection 3.3.2), floods (Chapter 3, Subsection 3.3.3), external and turbine-generated missiles (Chapter 3, Subsection 3.3.5 and Chapter 10 Subsection 10.2.3), and earthquakes (Chapter 3, Subsections 3.3.1). The Reactor Building protects the fuel pool, liner, and fuel storage racks from these hazards.

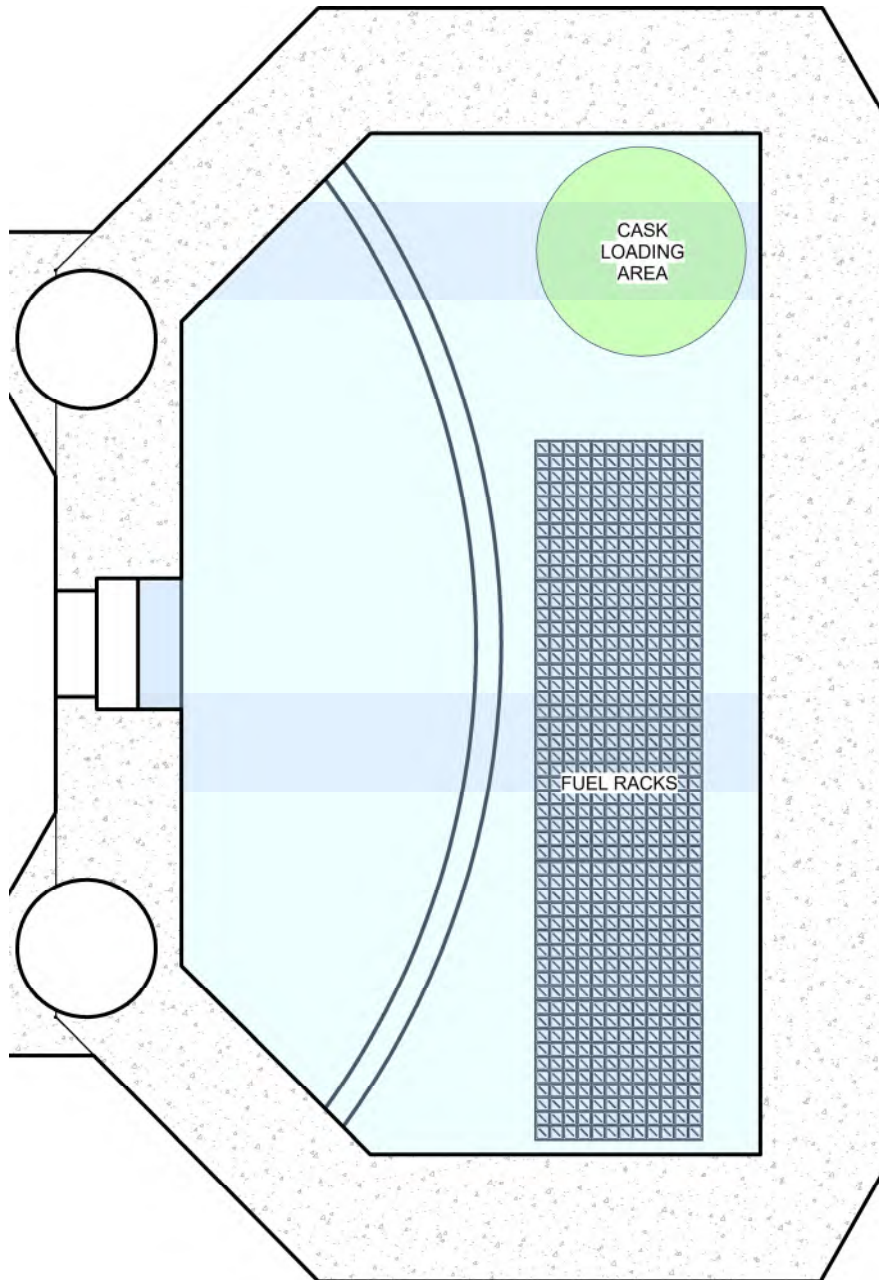
The design of the fuel storage racks is such that  $K_{eff}$  remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Drop of a fuel assembly onto fuel assemblies stored in the fuel pool is discussed in Chapter 15, Subsection 15.5.8. Handling equipment capable of carrying loads heavier than fuel components are prevented by design and administrative controls from carrying loads over the fuel pool (Subsection 9A.8).

#### **9A.1.2.11 References**

- 9A.1.2-1 CNSC Regulatory Document REGDOC-2.4.3, "Nuclear Criticality Safety."
- 9A.1.2-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.1.2-3 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.1.2-4 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."
- 9A.1.2-5 ANSI 57.1, "Design Requirements for Light Water Reactor Fuel Handling System," American National Standards Institute.



**Figure 9A.1.2-1: Fuel Pool Arrangement**

### **9A.1.3 Fuel Pool Cooling and Cleanup System**

The primary function of the FPC is to provide continuous cooling of the water volume in the fuel pool to remove decay energy from spent fuel, and to provide replacement coolant inventory from a variety of sources, both to ensure spent fuel is kept cool and submerged throughout the life of the plant. In addition, FPC includes demineralization and particulate filtration to maintain coolant quality and to reduce general area dose. FPC can be realigned to provide cooling and cleanup to the reactor cavity and equipment pools as necessary.

The FPC system is generally classified as Defense Line 2, Safety Class 3 (DL2/SC3) system.

The portion of the system that provides makeup capacity to the pool is classified as Defense Line 4b, Safety Class 3 system.

#### **9A.1.3.1 System and Equipment Functions**

System and equipment functions associated with the FPC system include the following.

##### **9A.1.3.1.1 Normal Functions (Non-Safety-Category)**

1. The FPC system maintains the water quality of the Fuel Pool, Reactor Cavity Pool, and Equipment Pool through filtration and demineralization during Modes 1-6. Refer to Chapter 16, Subsection 16.7 for definitions of operating Modes.

##### **9A.1.3.1.2 Normal Functions (Safety-Category)**

1. The FPC system provides cooling (SC3) of the Fuel Pool, Reactor Cavity Pool, and Equipment Pool during Modes 1-6.
2. The FPC system provides makeup capacity for the Fuel Pool (SC3), Reactor Cavity Pool, and Equipment pool (SC3) during Modes 1-6 delivered directly to the surge tanks and circulated to the pools.
3. The FPC system maintains the water level of the Fuel Pool (SC3) for shielding and cooling.
4. The FPC system provides high heat load cooling of the Fuel Pool and Reactor Cavity Pool during Mode 6 (refueling outage).

##### **9A.1.3.1.3 Off-Normal Functions (Non-Safety-Category)**

1. The FPC system is able to restore the Fuel Pool temperature to normal operating limits from elevated temperatures due to an off-normal event upon restoration of the forced cooling components of the system.

##### **9A.1.3.1.4 Off-Normal Functions (Safety-Category)**

1. The FPC system provides makeup capacity for the Fuel Pool (SC3) during off-normal events, independent of the forced cooling portion of the system.
2. The FPC system maintains the water level of the Fuel Pool (SC3) for shielding and cooling during off-normal events.

#### **9A.1.3.2 Safety Design Bases**

The FPC System provides continuous cooling normal plant operations, and makeup capacity for the fuel pool during normal and off-normal plant operations. Shielding is afforded by maintaining fuel pool water levels. The FPC System provides high heat load cooling of the Fuel Pool and Reactor Cavity Pool during Mode 6 (refueling outage).

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

The fuel pool and its supporting cooling system functions are capable of maintaining the fuel pool temperature within a set upper limit given the:

1. Normal decay heat load that occurs when an accumulation of spent fuel equal to 8 years is in the fuel pool, with the newest batch having just been placed in the pool during refueling within the first 72 hours after shutdown
2. Maximum decay heat load, which is the normal heat load plus the addition of the decay heat from a full core that is off-loaded to the fuel pool 72 hours after shutdown

FPC System provides make up capacity to the fuel pool during off normal events to maintain adequate fuel coverage.

The design of the FPC system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.12.2 (Reference 9A.1.3-1) as related to the handling and storage systems associated with spent fuel specifically heat removal in operational states. Refer to Subsection 9A.1.4 for information related to the handling of spent fuel and Subsection 9A.1.2 for information related to spent fuel storage systems.

### **9A.1.3.3 Description**

Refer to Figure 9A.1.3-1 which depicts the FPC.

The BWRX-300 includes capabilities for the handling and storage of new and spent fuel to support operation of the plant with fuel cycle durations from 12 to 24 months. The FPC system provides cooling and cleaning of the water in the Fuel Pool, reactor cavity pool, and equipment pool during power operations.

The FPC system and its supporting cooling system functions are capable of maintaining the Fuel Pool temperature within a set upper limit as described in Subsection 9A.1.3.2.

The FPC System provides make up capacity to the Fuel Pool during Off-Normal events to maintain adequate fuel coverage. The FPC system is constructed such that any breach of the system cannot cause draining of the Fuel Pool to a level below 3.05 m above the top of active fuel.

The FPC System consists of two trains of equipment, each with a pump, demineralizer, and heat exchanger. The capability to bypass the demineralizer while providing active cooling to support restoration of the pool temperature from conditions exceeding the demineralizer operational temperature limits is provided. Each set of components are placed in parallel to provide single train operation and cross connecting of trains should a component fail. A single train is sufficient to prevent bulk boiling in the Fuel Pool. If both trains are rendered inoperable, the Fuel Pool is sized such that it can retain sufficient coverage of the fuel for seven days, and FPC can provide makeup capacity independently of the forced cooling trains.

#### **9A.1.3.3.1 Component Description**

The following information is provided relative to the major equipment and components in the FPC.

#### **Surge Tank Description**

The surge tank(s) receive inventory from the reactor cavity and fuel pool as it overflows weirs located near the top of the reactor cavity and fuel pool. Surge tank(s) are utilized to provide protection from a breach in the system draining either pool volume (versus direct suction) and provide a dampening effect for volume changes in the pools due to outage or dry cask evolutions. The primary advantage of a separate surge volume is that the pool level can be constant, while natural evaporation or addition or removal of submerged equipment only varies the level of the

surge tank. Coolant inventory addition can be added directly to the surge tank during normal operating conditions. The FPC system surge tanks meet the requirements of American Society of Mechanical Engineers (ASME) Section III NCA and NCD.

### **Pumps**

Each pump represents the beginning of two separate trains of the FPC system. They can be run individually during normal heat removal and cleanup modes (A1, A2) or in combination for high heat load (B, C) operating modes. Pumps are sized for maximum efficiency at the normal operating point. Pump internal recirculation cavitation is avoided within the entire range of operation. The pump selection, installation and system design assure that the minimum available Net Positive Suction Head (NPSH) and minimum submergence meet the Hydraulic Standards Institute guidelines for all steady-state and transient conditions of operation. If parallel operation is specified, the head rise from rated point to shutoff is at least 10%.

The FPC system pumps meet the requirements of ASME Section III NCA and NCD.

### **Heat Exchangers**

Each train includes two heat exchangers (four in total) used to reject waste heat to the Plant Cooling Water System (PCW) (Subsection 9A.2.1). Each heat exchanger is a shell and tube design, with reactor coolant flowing on the tube side to take advantage of inherent shielding and reduce dose. Each heat exchanger is sized to fit within the Reactor Building hatch to allow removal or replacement during the operating life and decommissioning of the plant. Heat exchangers are designed, manufactured, installed, and tested in accordance with applicable codes and standards, including Thermal Exchanger Manufacturers Association, API-662, ASME Boiler and Pressure Vessel Code (BPVC), Section III and ASME BPVC Section VIII. Large shell and tube heat exchangers are designed with provisions for either tube bundle replacement or individual tube replacement and individual tube plugging in place. Heat exchanger fouling factors are established considering conservative predictions of material buildup based on actual system and equipment designs and expected plant operating conditions. Heat exchanger tubes are seamless.

### **Piping/Valving**

Piping in radioactive systems such as the FPC System have butt-welded connections, rather than socket welds, to reduce crud traps. Features to prevent flow discontinuities that can lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated into the design. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout. Mitigating engineering features are added where avoidance is not possible. These piping and valve design features reflect implementation of ALARA guidance as presented in Chapter 12, Section 12.1.

FPC system piping potentially containing resin is designed to be continuously sloped downward to the receiving system or tank.

The FPC system piping, and valves meet the requirements of ASME Section III NCA and NCD.

### **Demineralizers / Filters**

Each demineralizer train of FPC system contains a particulate filter and deep bed mixed resin demineralizer which can each support 100% of the train's flow. The filter may be backwashed to the Liquid Waste Management System (Chapter 11, Section 11.2) to remove accumulation and reduce dose. The demineralizers each operate in both normal (A1, A2) and high heat (B) modes, but can also be bypassed if the temperature of the system coolant exceeds the operational limits of the demineralizer (C).



Process equipment that accumulates a radiation source from filtering process streams such as the Condensate Filters and Demineralizers System (CFD) (Chapter 10, Subsection 10.3.1), and the FPC systems are remotely operated, including the backwashing operations. Provisions are made for remotely backflushing the filters and demineralizers. FPC system filters are backwashed into a backwash receiving tank, which then is routed to the Radwaste Systems. All FPC System valves (e.g., inlet, outlet, recycle, vent, and drain) on the filters and demineralizers are located outside the shielded cubicles in a separate shielded cubicle or area together with associated piping, headers, and instrumentation.

The FPC system filter element shell and demineralizer vessel meet the requirements of ASME Section III NCA and NCD.

### **Off-Normal Makeup**

The Off Normal makeup portion of the system represents the seismic DL4b portion of the system intended to provide replacement inventory to the fuel pool. It utilizes multiple sources to maintain pool level in the event of a failure of the active cooling portion of FPC.

#### **9A.1.3.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from Intergranular Stress Corrosion Cracking (IGSCC) (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

Features to prevent flow discontinuities that can lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated into the design. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout.

#### **9A.1.3.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.1.3-1 for FPC interfaces with other equipment or systems.

#### **9A.1.3.6 System and Equipment Operation**

The FPC system is primarily intended to maintain pool temperatures during plant operation and refuel outages. During Modes 1-5, a single train of FPC is running to provide cooling and cleanup of the Fuel Pool, Reactor Cavity, and Equipment Pool inventory (Mode A1 and A2.) Supply to each pool may be balanced based on temperature variations and heat loadings in each volume. During refuel outages (Mode 6) the FPC system may be run in normal (FPC mode A1 or A2) or High Heat Mode (FPC mode B) as the temperature of the Reactor Cavity and Fuel Pool dictates. In the event of elevated pool temperatures, which exceed the operating specifications of the demineralizers such as restoration from an Off-Normal event, the FPC system may be run in High Heat – Bypass (FPC mode C) which utilizes both equipment trains of pumps and heat exchangers to provide maximum cooling without cleanup. Once the pool temperature returns to acceptable levels, the FPC system may then return to Mode B or Mode A as required.

##### **9A.1.3.6.1 Initial Configuration (Pre-Startup)**

As a system intended to always be in operation, there are no specific pre-startup conditions. Operation of the FPC system is independent of reactor operation.

##### **9A.1.3.6.2 System Startup**

Generally, a single train of FPC is running continuously.

During startup of a train, confirmation of the following is obtained:

- Valve alignment for the intended mode
- Sufficient surge capacity
- Pump inlet pressure to avoid cavitation
- Pump start
- Anticipated component performance indications

This same process may be used to add a second train during high heat modes. In the changeover from one train to the other, it is preferable to start the second train and confirm normal operation prior to shutting down the first train.

#### **9A.1.3.6.3 Normal Operations**

During normal operations the FPC system operates in the following modes:

##### **Mode A – Normal Heat Load**

The most frequent mode of operation of the FPC system is “normal heat load” operation, intended to be used during plant operating Modes 1-5. This mode includes a single train of the FPC system operating to remove heat generated by spent fuel stored in the fuel pool, as well as any thermal contribution from the Passive Containment Cooling System (Chapter 6, Subsection 6.3.3) to the equipment pool, or thermal leakage from containment to the reactor cavity pool. Cleanup consisting of particulate filtration and demineralization of 100% of the water passing through the system is performed. During Mode A, coolant inventory can be added from the condensate storage tank as required to makeup evaporation losses or over-boarded to the condensate storage tank after cleanup and cooling to create additional pool volume for submersion of spent fuel dry cask equipment. Mode “A1” and “A2” may be used to signify which train is in operation, both trains function identically.

##### **Mode B – High Heat Load**

Mode B utilizes both trains of FPC equipment running simultaneously to double the volume of water cooled and cleaned. This mode still retains 100% filtration and demineralization and is intended to be used predominately during plant operating Mode 6 “Refueling,” when high activity spent fuel is transferred to the fuel pool, and water clarity and dose are of particular concern. As with Mode A, coolant inventory can be added or overboarded to support outage operations and maintain water level.

#### **9A.1.3.6.4 Off-Normal Operations**

##### **Mode C – High Head Load – Bypass**

Mode C functions identically to Mode B, except the demineralizers are isolated from the system and a bypass is utilized. Mixed bed resin demineralizers tend to have modest thermal operating limits, typically less than 60 °C which when exceeded can cause resin excursions contaminating plant equipment and pools. To retain the high heat removal capacity but prevent damage to the resin beds during high pool temperature events, Mode C may be utilized until the water temperature is restored to a threshold allowing Mode B service.

##### **Mode D – Active Cooling Inoperative**

Mode D is an off-normal condition where the active cooling and cleanup portion of the FPC system is unable to be operated such as during a site blackout. Mode D operations allows for multiple sources of coolant addition directly to the Fuel Pool, bypassing the remaining system or any potential breeches that may have occurred. The plant sources include the condensate storage

tank, and the Fire Protection system. Mode D also includes an Emergency Mitigating Equipment (EME) connection allowing addition of coolant through temporary means. The Fuel Pool volume is intended to contain sufficient water inventory to allow seven days of spent fuel decay heat to be absorbed while maintaining sufficient coverage of 3.05 m over fuel. Operations in Mode D is required for additional inventory to be added to meet the 30-day event criteria.

#### **9A.1.3.6.5 System Shutdown**

The FPC system is intended to operate, with at least a single train, during all modes of plant operation. Individual trains may be idled or placed in standby through shutdown of pumps and isolation of equipment as heat load or maintenance and testing require.

#### **9A.1.3.7 Instrumentation and Control**

As an SC3 system, control of the FPC system is performed through the Safety Class 2 and 3 Instrumentation and Control System indication of pool temperature and water level. During normal operation, pumped capacity is varied through use of one or both system trains, with output balanced through motor operated control valves to the equipment, reactor cavity, and fuel pool to distribute cooled water in a manner best matching the sources of thermal energy. Addition or reduction of inventory, both as a natural function of environmental conditions, as well as supporting maintenance and outage evolutions, can be performed manually.

During off-normal events, with the forced cooling portion of the system assumed to be inoperable, make up capacity can be initiated from various sources, again through the Safety Class 2 and 3 Instrumentation and Control System, as redundant Fuel Pool level indication requires. Major components (pumps, demineralizers, heat exchangers) have performance indication through the Safety Class 2 and 3 Instrumentation and Control System, and component protection such as bypassing of demineralizers if system temperature exceeds limits and stopping of pumps on low flow and low suction pressure signals are automated.

#### **9A.1.3.8 Monitoring, Inspection, Testing, and Maintenance**

Maintenance and testing support equipment reliability. SSC are designed to facilitate operation and maintenance. Trending of operational characteristics of equipment such as demineralizers, filters, and heat exchangers identifies reductions in performance signaling maintenance is required. Trains are switched periodically to balance equipment usage and provide surveillance opportunity to both trains. In-Service Inspection (ISI) and In-Service Testing (IST) requirements are established and include inspection/test frequency for SSC.

Maintenance activities related to the FPC system fall into two categories:

1. Ongoing/Frequent Maintenance: Anticipated maintenance activities such as backflushing of particulate filters, flushing and replacement of spent resin in demineralizers, and decontamination of lines to reduce area dose.
2. Infrequent Maintenance: Anticipated activities that occur during the life of the plant which includes equipment replacement of components such as pump impellers, motors, valves, heat exchangers, and filter cartridges. These maintenance evolutions take place as a response to long-term trending of system performance or as preventive maintenance scheduled based on anticipated component life span.

In both cases, maintenance on the FPC system is performed predominately during Plant Mode 1 operation, when thermal loads in the Fuel Pool are low allowing a single operable train to satisfy plant needs while the other may be isolated for work.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

#### **9A.1.3.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. Chapter 12, Subsection 12.3.8.3 presents Fuel Pool Cooling major ALARA design considerations.

#### **9A.1.3.10 Performance and Safety Evaluation**

The FPC System is designed to perform its function in a reliable and failure tolerant manner. This reliability is achieved with the use of rugged and redundant equipment. Each set of components (pumps, demineralizers, heat exchangers) are placed in parallel to provide single train operation and cross connecting of trains should a component fail. A single train is sufficient to prevent bulk boiling in the fuel pool. If both trains are rendered inoperable, the fuel pool is sized such that it can retain sufficient coverage of the fuel for seven days, and FPC can provide makeup capacity independently of the forced cooling trains.

The FPC System Safety-Category functions during Normal and Off-Normal conditions include providing makeup capacity to the Fuel Pool, maintaining the water level of the Fuel Pool for shielding and providing Fuel Pool Cooling. In addition, Off-Normal makeup piping and valves which are SC3 and Seismic Category A are provided to allow remote addition of water to the Fuel Pool during Off-Normal conditions through temporary means thereby ensuring spent fuel is cooled and fuel pool water levels are maintained.

In the event of an aircraft impact, there are provisions to refill the pool utilizing Emergency Mitigating Equipment (EME) or other equipment and water sources in a suitable time frame such that adequate cooling of the spent fuel is maintained based on keeping the fuel covered with sufficient water.

The Fuel Pool is designed to dissipate the maximum spent fuel decay heat through heat-up and boiling of the fuel pool water. The most conservative heat load for the Fuel Pool occurs when the Fuel Pool contains spent fuel from the normal decay heat load that occurs when an accumulation of spent fuel equal to 8 years is in the fuel pool, with the newest batch having just been placed in the pool during refueling within the first 72 hours after shutdown.

The Fuel Pool is located in the Reactor Building which provides protection against natural phenomena; supporting the ability of the FPC System to perform its Safety-Category functions.

#### **9A.1.3.11 References**

- 9A.1.3-1      CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

**Table 9A.1.3-1: Fuel Pool Cooling and Cleanup System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Reactor Building, Fuel Pool, Reactor Cavity, and Equipment Pool	Overflow from the Fuel Pool and Reactor Cavity Pool weirs enters skimmer surge tanks and is pumped through cleanup and heat removal subsystems, then returned to the bottom of the Equipment Pool, Reactor Cavity Pool, and Fuel Pool as required.	Boundary from the Reactor Building exists at overflow from pools to Skimmer Surge Tanks. Boundary at supply exists at sparger nozzle(s) submerged in pools.
Liquid Waste Management System (LWM)	Condensate Storage Tank provides additional coolant inventory or storage volume for overboarding to reduce inventory. LWM also supplies liquid for back wash and flushing of cleanup systems.	Boundary exists at first isolation valve at each connection point.
Solid Waste Management System (SWM)	Repository for spent resin from demineralizers and backwash of filter elements. Flushed via piping to SWM using LWM inventory.	Boundary exists at first isolation valve at each connection point.
Offgas System (OGS)	Vent used to allow compressed air circulation of demineralizer resin and venting of volume.	Boundary exists at first isolation valve at each connection point.
Plant Cooling Water System	PCW supply for heat exchangers.	Boundary exists at first isolation valve at each connection point.
Plant Pneumatics System (PPS)	Air supply to allow mixing of demineralizer resin. Air Operated Valve operation as required.	Boundary exists at first isolation valve at each connection point.
Safety Class 2 and 3 Instrumentation and Control	Provides instrumentation and control for Safety Class 3 functions of FPC system.	Boundary exists at each rack prior to multiplexer.
Safety Class 2 and 3 Electrical Distribution System	Electricity supplied to pumps, motorized valves, and instrumentation/control throughout system.	Boundary located at individual component sub feed.
Process Radiation and Environmental Monitoring System (PREMS)	PREMS provides area radiation detector near the demineralizer, process sampling for radiological analysis and chemistry control.	Area radiation detector, process sampling routed to sample station.
Fire Protection System	Provides additional coolant inventory in Off-Normal conditions.	Boundary exists at first isolation valve at each connection point.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
EME	Although not a standalone system, a connection is included to provide makeup capacity from remote location.	Boundary at final valve of EME connection point.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

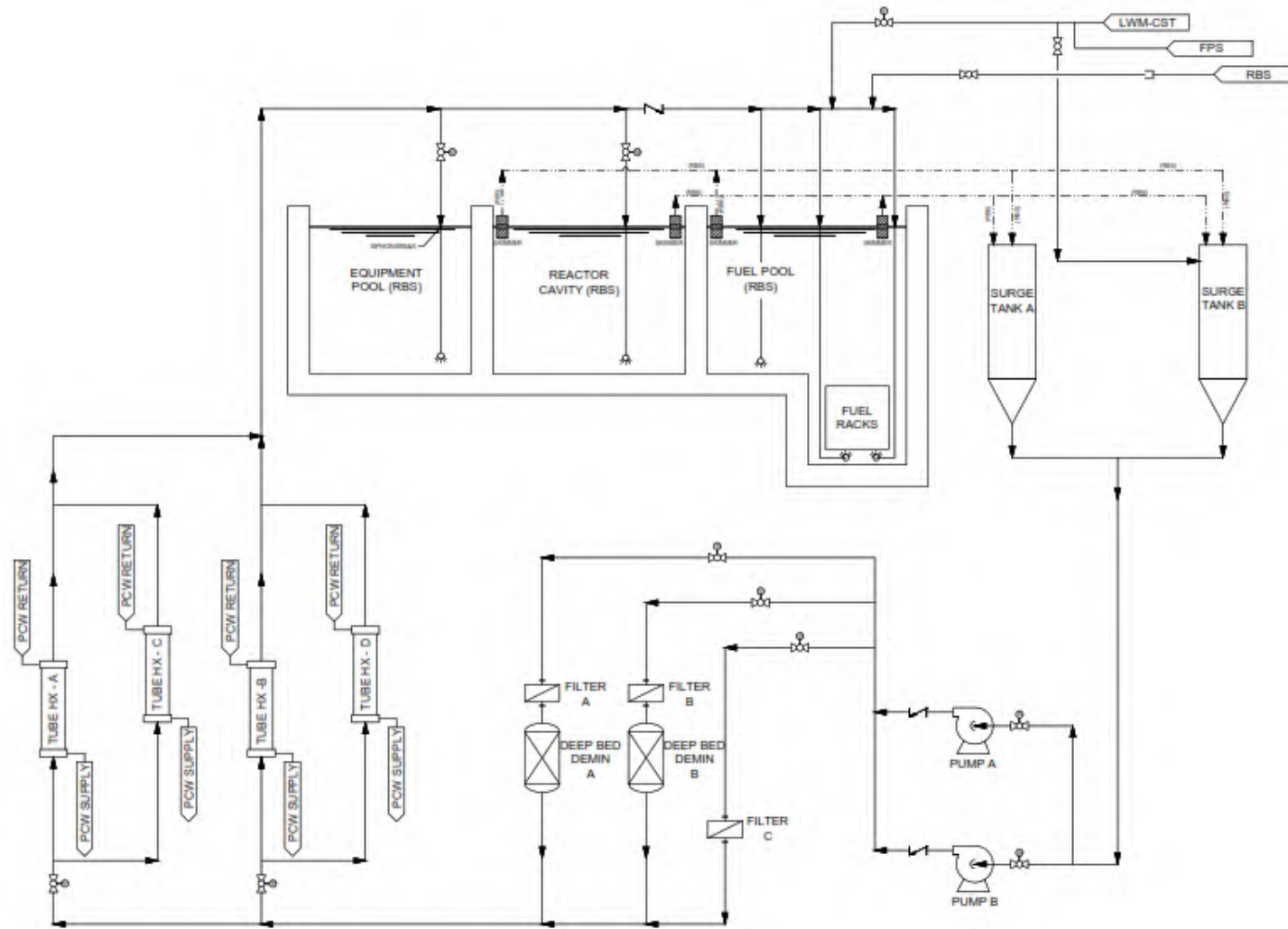


Figure 9A.1.3-1: Fuel Pool Cooling and Cleanup System

#### **9A.1.4 Handling Systems for Fuel Cask Loading**

##### **9A.1.4.1 System and Equipment Functions**

The system and equipment functions associated with the handling system for fuel cask loading, provide the means to transfer spent fuel located in the fuel pool into a spent fuel storage cask and transport of the spent fuel storage cask to a location for long-term storage. The following information is provided relative to demonstrating compliance to the requirements CNSC REGDOC-2.5.2 Section 8.12.1 (Reference 9A.1.4-1) as it pertains to fuel cask loading.

##### **9A.1.4.2 Safety Design Bases**

1. Fuel handling devices have provisions to avoid dropping or jamming of spent fuel assemblies during transfer operations.
2. Handling equipment used to raise and lower spent fuel has a limited maximum lift height so that the minimum required depth of water shielding is maintained.
3. Criticality during fuel handling operations is prevented by maintaining a geometrically safe configuration throughout the spent fuel transfer operation. Operators follow a strict movement plan and procedure and that includes pick up and set down locations for each step and those are followed and verified concurrently as each step is started and completed.
4. In the event of a Safe Shutdown Earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of Safety Category 1 SSC.
5. Physical safety features are provided for personnel who operate handling equipment.

Refer to Subsection 9A.1.2.2 for the Safety Design Bases associated with the Fuel Pool. Refer to Section 9A.8 for the Safety Design Bases associated with the Cranes, Hoist, and Elevator System.

##### **9A.1.4.3 Description**

###### **9A.1.4.3.1 Refueling Platform**

The Refueling Platform is a gantry-type crane that spans the reactor vessel cavity, equipment pool, and fuel pool and is employed to handle fuel and perform other ancillary tasks in the reactor building. It is equipped with a traversing trolley on which is mounted a telescoping mast and fuel grapple. A monorail and auxiliary hoist are also provided. The Refueling Platform is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refueling process. The Refueling Platform does not perform a Safety-Category function but is designed as Seismic Category I.

###### **9A.1.4.3.2 Irradiated Fuel Canister Loading**

The spent fuel is stored in the Fuel Pool before being loaded into a canister. Typical loading of a canister includes the following. Loading of the canister is performed in the Fuel Pool. The canister containing the spent fuel is then loaded into a concrete overpack. Canister loading patterns are determined by the age of fuel, exposure, and decay heat. A campaign normally involves loading the spent fuel canister while the unit is on-line.

An empty canister placed in a transfer cask for shielding is moved into the fuel pool. The selected bundles are moved under water from the spent fuel racks to the allotted location in the spent fuel canister. Once a spent fuel canister is filled with spent fuel, a video recording of the bundle serial numbers is performed, and the lid is placed on the canister. The canister is lifted from the fuel pool, drained, rinsed, and set on the cask pad on the refueling floor where the canister is



decontaminated. The canister is vacuum dried and sealed by welding the lid using remote welding techniques to minimize radiation exposure to workers. The canisters are filled and slightly pressurized with an inert gas, such as helium. The sealed canisters are leak tested and a non-destructive examination is performed on the weld. The shielded canister is installed into an overpack and then fitted with an overpack lid and placed in long-term storage.

Each spent fuel canister may contain several damaged fuel bundles; however, the specific loading is vendor dependent. The failed fuel bundles can stay in the fuel pool indefinitely without any special controls. If desired, fuel bundles can be reconstituted with new fuel or dummy rods replacing damaged rods. This is done on the fuel inspection stands in the fuel pool with long handled tools.

#### **9A.1.4.3.3 Irradiated Fuel Canister Movement to ISFSI Pad**

Typical canister movement from the fuel pool to the Independent Spent Fuel Storage Installation (ISFSI) pad is performed under the guidance and participation of Security and Health Physics. The RB Polar Crane is used to lift the transfer cask to the refuel floor and subsequently to the truck bay. The canister is transferred from the transfer cask to the storage cask and placed in long-term storage.

#### **9A.1.4.3.4 ISFSI Pad**

The ISFSI pad size is related to the vendor selected for the design and manufacturer of the spent fuel casks and the number of fuel assemblies to be stored.

The land usage required to provide long-term dry storage of spent fuel includes the fenced off area necessary to provide an acceptable radiation protection and security zone. Perimeter fencing, intrusion detection system, lighting and cameras are provided as required by CNSC irradiated fuel storage facility requirements.

#### **9A.1.4.4 Materials**

Materials selected for use in the Handling Systems for Fuel Cask Loading are chosen based upon the operating conditions to which they are required to function. The fuel storage racks use surveillance coupons for monitoring and evaluation of neutron absorber material.

#### **9A.1.4.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.1.4-1 for Handling System for Fuel Cask Loading interfaces with other equipment or systems.

#### **9A.1.4.6 System and Equipment Operation**

The handling of new fuel is described in Section 5 of the Safeguards Annex. The handling of irradiated fuel inside the fuel pool is described in Subsection 9A.1.2. Handling of irradiated fuel from the fuel pool to a storage canister is discussed in Section 5.1 of the Safeguards Annex. Handling of the spent fuel storage canister with respect to the transfer of the spent fuel canister to an overpack is discussed in Subsection 9A.1.4.3. Handling of the overpack during transport to long-term storage is discussed in Subsection 9A.1.4.3.

#### **9A.1.4.7 Instrumentation and Control**

Refer to Subsection 9A.1.3.7 for information pertaining to instrumentation and control associated with the fuel pool and Section 9A.8 for information pertaining to instrumentation and control associated with the RB Polar Crane.

#### **9A.1.4.8 Monitoring, Inspection, Testing, and Maintenance**

Maintenance of refueling equipment and tooling is performed prior to an outage to promote optimum reliability.

Refer to Subsection 9A.1.2.8 for information pertaining to the Monitoring, Inspection, Testing, and Maintenance of the fuel pool. Refer to Subsection 9A.8.1.8 for information pertaining to the Monitoring, Inspection, Testing, and Maintenance of Cranes, Hoists and Elevators.

#### **9A.1.4.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.1.4.10 Performance and Safety Evaluation**

The handling of new fuel is described in Section 5 of the Safeguards Annex. The handling of irradiated fuel inside the fuel pool is described in Subsection 9A.1.2. Handling of irradiated fuel from the fuel pool to a storage canister is discussed in SubSection 9A.1.4.3.2. Handling of the spent fuel storage canister with respect to the transfer of the spent fuel canister to an overpack and transport to the Independent Spent Fuel Storage Installation is discussed in Subsection 9A.1.4.3.3:

1. The fuel handling equipment is designed such that probability of dropping a fuel assembly following a safe shutdown earthquake has been minimized.
2. Each fuel assembly and control rod is placed strategically to maintain a non-critical configuration by following a specific fuel movement plan.
3. Underwater transfer of spent fuel assemblies provides radiation shielding. The fuel handling equipment has provisions to limit maximum height to maintain sufficient water inventory above the top of the fuel assembly.
4. The fuel handling equipment includes controls and interlocks that impose limits upon system operations, ensuring clearance between structures, systems, and components, thereby preventing the potential for mechanical damage to fuel during fuel transfer operations.

#### **9A.1.4.11 References**

- 9A.1.4-1    CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.1.4-1 Handling System for Fuel Cask Loading Interface**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Fuel Pool Cooling and Cleanup System	FPC provides water clarity for underwater visibility FPC also provides water cooling to support fuel pool cooling activities.	The Handling System for Fuel Cask Loading equipment is immersed in water treated and cooled by FPC.
Core and Fuel	Supports the transfer of fuel assemblies in and out of the reactor core.	The mast of the Refueling Platform engages the bail handle of the fuel assembly.
Plant Pneumatics System	PPS provides pressurized air for powering pneumatic motors and tooling.	Gate seals, nozzle plugs, and tooling.
Non-Safety Electrical Distribution System	Provides electricity to the Handling System for Fuel Cask Loading equipment.	Inspection tooling, underwater lights, refueling platform, auxiliary platform, and tooling.
Grounding and Lightning Protection System	Provides electrical grounding for equipment.	Equipment and inspection equipment.
Cranes, Hoists and Elevators	The RB Polar Crane is used during the disassembly and re-assembly of the Reactor Pressure Vessel.	The Handling System for Fuel Cask Loading equipment couples to RB Polar Crane hook.
Heating Ventilation and Cooling System	Provides control of heat and humidity to work areas.	Ambient condition suitable work.
Reactor Building Structure	Supports the loads of equipment and provides railing for traversing of the Refueling Platform.	Equipment either rests on the refuel floor or its weight is transmitted to the RB walls through the RB Polar crane.

## **9A.2 Water Systems**

### **9A.2.1 Plant Cooling Water System**

The PCW system provides cooling water to Non-Safety and Safety Class 3 components and provides a barrier against radioactive contamination of the Circulating Water System (CWS) (Chapter 10, Section 10.8). It consists of two piping subsystems, Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution, that provide cooling water to various heat exchangers.

The safety classification of the PCW as well as interfacing Structures, Systems, and Components (SSC) is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.2.1-1).

#### **9A.2.1.1 System and Equipment Functions**

##### **9A.2.1.1.1 Normal Functions (Non-Safety-Category)**

The PCW circulates cooling water to the Reactor Component Cooling Water Piping Distribution and the Turbine Component Cooling Water Piping Distribution during normal operation and during anticipated operational occurrences, including startup, power operation, hot shutdown, cold shutdown, stable shutdown and refueling. The PCW is responsible for rejecting the total heat load associated with the equipment coolers in the Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution.

PCW operation is normally automatic based on plant operational mode but can be started or stopped manually from the Main Control Room and can operate at any time, regardless of the operational status of the generating unit.

The PCW design supports the redundant Reactor Component Cooling Water Piping Distribution and single Turbine Component Cooling Water Piping Distribution.

The cooling water temperature is maintained by a temperature control valve arrangement that bypasses the PCW heat exchangers with a portion of the return water.

##### **9A.2.1.1.2 Normal Functions (Safety-Category)**

The system provides the following normal safety functions:

- Control of the fuel pool temperature
- The system provides cooling for the SDC heat exchanger
- A barrier against radioactive contamination of the CWS

##### **9A.2.1.1.3 Off-Normal Functions (Safety-Category)**

Upon a Loss-of-Offsite Power (LOOP), the operating PCW pump(s) will trip. The pumps are automatically repowered from the standby diesel generators.

In the event of LOOP, the reactor cooling loops are used to support cooling of the Fuel Pool and Reactor to bring the plant to cold shutdown condition in 36 hours if necessary assuming the most limiting single active failure (single train in use), with use of the Isolation Condenser System (ICS) (Chapter 6, Section 6.2). If both trains are still in use, plant systems provide adequate shutdown cooling capacity to achieve appropriate reactor coolant temperature in 24 hours.

##### **9A.2.1.1.4 Off-Normal Functions (Non-Safety-Category)**

The system does not perform any Safety-Category functions during off-normal conditions.

#### **9A.2.1.2 Safety Design Bases**

The Safety Design Bases for the PCW includes the following Safety Class 3 functions:

- Temperature control of the fuel pool
- Cooling for the SDC heat exchanger

#### **9A.2.1.3 Description**

Figure 9A.2.1-1 depicts the Plant Cooling Water System.

The PCW system consists of two trains, each containing one pump and one heat exchanger, that handle the Reactor Component and Turbine Component cooling loads. One train is normally in operation while the other is on standby, to be started in event the operating train needs to be shutdown.

These independent trains are cross connected using manual cross ties to allow for online maintenance. If necessary, the Turbine Component Cooling Water Piping Distribution can be isolated from each Reactor Component Cooling Water Piping Distribution by closing the supply and return header valves on each train, and each Reactor Component Cooling Water train operates independent of the other.

Cooling water supplied by the PCW is continuously circulated through various auxiliary equipment heat exchangers and rejects the heat transferred to the CWS. The Turbine Component Cooling Water Piping Distribution is a single piping distribution that serves all equipment in the Turbine Building that do not support reactor cooling functions. The Reactor Component Cooling Water Piping Distribution consists of two redundant piping distributions that support all Reactor Building equipment cooling functions, Plant Pneumatic System and any equipment located outside the Reactor Building associated with reactor cooling activities.

Although PCW supports all plant equipment cooling functions, the design redundancy and isolation capability is centered around the ability to provide redundant cooling supply to nuclear systems.

One train of PCW equipment is normally in operation and in the event that train is lost, the standby train is started. During plant shutdown both trains of cooling equipment are used to provide required flow for Turbine Building and Reactor Building cooling functions. The Reactor Component Cooling Water Piping Distribution cools redundant equipment associated with nuclear cooling functions, thus if one train of the Reactor Component Cooling Water Piping Distribution is lost all redundant equipment will operate on the remaining train.

The PCW pumps and heat exchangers are in the Turbine Building. The pumps in each train are powered from separate busses. During a LOOP, the pumps are powered from the two Safety Class 3 standby diesel generators.

The PCW utilizes plate and frame type heat exchangers. This design mitigates cross-contamination of either PCW or CWS.

Temperature control valves in the system are provided to maintain the cooling water temperature within an allowable range.

Surge tanks provide a constant pump suction head and allow for thermal expansion of the PCW inventory. Makeup to the PCW inventory is from the PCW surge tank which is supplied from the Makeup Water System (Subsection 9A.9.5.1) through an automatic level control valve.

A chemical addition tank tie-in allows for manual introduction of corrosion inhibitor and pH control chemicals into the system.

The Turbine Building cooling loop serves but is not limited to the following equipment:

- Feedwater Pump Motors
- Feedwater Pump Variable Frequency Drive
- Condenser Pump Motor
- Vacuum Pump Skid
- Generator Cooler
- Isophase Cooler
- Lube Oil Cooler
- Electro-Hydraulic Control (EHC) Cooler

The redundant Reactor Building cooling loops serve the following equipment:

- Fuel Pool Cooling and Cleanup Heat Exchanger
- Isolation Condenser System Pool Cooling and Cleanup System (ICC) Heat Exchanger
- Shutdown Cooling System (SDC) Heat Exchanger
- Plant Pneumatics System Cooler

PCW equipment is designed for the plant normal operating environmental conditions specified for its location within the Turbine Building and Reactor Building. Refer to Chapter 3, Section 3.9 for information pertaining to Environmental Qualification of BWRX-300 SSC's.

#### **9A.2.1.3.1 Component Description**

##### **PCW Pumps**

The PCW pumps are designed to meet the requirements of ANSI/HI 1.3 (Reference 9A.2.1-2).

The pumps are constant speed, electric motor driven, horizontal centrifugal pumps. Pump impellers are less than the maximum diameter available for the pump casing. The pumps' mechanical seals minimize the potential for leakage and reduce the need to perform maintenance on shaft seal packings.

##### **PCW Heat Exchangers**

The PCW heat exchangers are designed to meet the requirements of ASME BPVC, Section VIII.

PCW is capable of performing design functions during all modes of operation.

Plate and frame heat exchangers are used to minimize the probability of possible cross-contamination between PCW and CWS. Strainers are installed on the raw water side of the exchangers due to the narrow passages plate and frame heat exchangers present. Leakage through holes or cracks in the plates is not considered credible based on industry experience with plate type heat exchangers. In addition, the heat exchangers are designed such that any gasket leakage from either PCW or CWS drains to the EFS (Subsection 9A.9.3).

The CWS water temperature on the other side of the heat exchanger to absorb the heat load. There is a bypass line provided for the PCW Heat Exchangers with an air-operated bypass temperature control valve. The flow of cooling water through the PCW Heat Exchangers is controlled by the PCW Heat Exchanger temperature control system aiming to achieve target

temperature based on the PCW Heat Exchangers outlet temperature readings sent to the Distributed control and Information System.

### **PCW Surge Tanks**

The PCW surge tanks are atmospheric corrosion-resistant tanks which are designed to meet the requirements of American Water Works Association D100. The surge tanks are sized to provide a pump suction head and to allow for thermal expansion. The surge tanks are located above the highest points in the PCW system. The makeup water supply is provided to maintain the water level in the tanks within the correct operating conditions.

### **Piping and Valves**

All Non-Safety-Category function piping is designed to meet the requirements of ASME B31.1, Power Piping.

Isolation valves are provided at the inlet and outlet of each heat exchanger and pump for maintenance and efficiency purposes. Isolation valves are provided at the interfaces with the components being cooled by PCW to allow for maintenance on those items without impact to the PCW system operation. These isolation valves add the ability to remove heat exchangers from the system cooling requirements without shutting down the entire PCW system.

For the PCW pump and heat exchanger train section, there are isolation valves around the pumps and heat exchangers. Each pump discharge line is provided with a check valve to prevent backflow through the pump.

Vents are located in high points and drains are located in all low points. This ensures that the system is completely filled with water and that there are no air pockets. Vents reduce the chance for water hammer after a pump start. Valve opening and closing times are selected to minimize water hammer effects.

Each flow path to the interface system heat exchangers is designed to have flow balancing features that may include fixed plate orifices and/or control or manual valves.

### **Chemical Pot Feeder**

The chemical pot feeder is designed to meet ASME BPVC, Section VIII requirements. The chemical pot feeder is used to manually add corrosion inhibitors, pH control chemicals, and biocides into the system on a periodic basis. Chemicals are compatible with radwaste processing equipment if PCW water becomes contaminated.

#### **9A.2.1.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

#### **9A.2.1.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.1-1 for PCW interfaces with other equipment or systems.

#### **9A.2.1.6 System and Equipment Operation**

##### **9A.2.1.6.1 Normal Operational Concept**

For normal start up, the PCW is filled through the surge tank. When surge tank levels reach low level, the Makeup Water System (Subsection 9A.9.5.1) refills the surge tank through an automatic level control valve.

For normal operation, one train which consists of one pump and one heat exchanger is in operation while the other train is on standby. If one heat exchanger fails, the standby heat exchanger has enough capacity to handle cooling requirements.

Plant cooling water flow through to heat exchangers and coolers throughout the Turbine Building and Reactor Building by way of the Turbine Closed Cooling Water Subsystem piping and Reactor Closed Cooling Water Subsystem piping.

The CWS water flows on the other side of the PCW main heat exchanger to absorb the heat load. There is a bypass line provided for the PCW Heat Exchangers with an air-operated bypass temperature control valve. The flow of cooling water through to the PCW Heat Exchangers is controlled by the PCW Heat Exchanger temperature control system based on the PCW Heat Exchangers outlet temperature.

#### **9A.2.1.6.2 Off-Normal Operational Concept**

For off-normal modes of operation, the PCW system (Reactor Component Cooling Water Piping Distribution) provides Defence-in-Depth (D-in-D) functions to remove decay heat from the reactor, fuel pool, and isolation condenser pools. PCW piping trains are isolated to provide fully independent cooling water loops to both trains of equipment. Both trains of PCW are capable of supporting D-in-D cooling requirements associated with the plant.

In the event of a loss of one train of the PCW cooling system equipment, the standby train is placed into service thereby precluding plant shutdown.

Upon a LOOP, the PCW pumps trip and restart upon power from the standby diesel generators.

Any leakage resulting in a low surge tank level signal, the standby train of PCW isolates the Reactor Component Cooling Water Piping Distribution line and automatically places any non-operating equipment in bypass. The Water, Gas and Chemical Pads (WGC) system provides makeup water to the surge tanks. The train that continues to lose water will eventually trip on low suction head, leaving at least one train of PCW operational to support nuclear D-in-D functions.

In cases of leak detection the two isolation valves responsible for the Reactor Component Cooling Water Piping Distribution train isolation automatically close and the standby train starts and proceeds with cooling of the redundant Reactor Component Cooling Water Piping Distribution coolers.

#### **9A.2.1.7 Instrumentation and Control**

In normal operation one cooling train is operating while being controlled by the Distributed Control and Information System in order to achieve the desired target temperature for each cooled line in the Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution. In events of failure of the operating train or pump discharge pressure drop, an automated process switches operations to the standby train.

Temperature is monitored downstream of each system heat exchanger and readings are reported back to the main control room to monitor the heat exchanger performance.

For designated heat exchangers, bypass and discharge temperature control valves operating in split range control mode are used to control temperature. Instrumentation Temperature monitoring is provided, but not limited to the following locations:

- At the outlets of all equipment coolers
- Downstream and upstream of each PCW heat exchanger

Pressure monitoring is provided, but not limited to the following locations:



- At the suction and discharge of each PCW pump
- Downstream of the PCW heat exchangers

Differential Pressure monitoring provided, but not limited to the following locations:

- Across each PCW heat exchanger
- Across equipment coolers
- Across the pump suction strainers

Flow monitoring provided, but not limited to the following locations:

- At the outlet of each PCW heat exchanger
- At the inlets of selected system heat exchangers as needed

Level monitoring is provided at the following locations:

- On each surge tank

Temperature Control Valves are provided at the following locations:

- Downstream and Across (bypass) each SDC heat exchanger
- Downstream and Across (bypass) the Generator Coolers
- Downstream and Across (bypass) the Lube Oil Coolers
- Downstream and Across (bypass) each PCW heat exchanger

#### **9A.2.1.7.1 Controls**

The PCW system is monitored from the Main Control Room. PCW controls and interlocks for the main components are described below:

1. Temperature monitoring is provided downstream of the PCW heat exchangers on the main cooling water supply line, that send a signal to the DCIS to control cooling water temperature by bypassing a portion of the water around the PCW heat exchangers. This is accomplished using an air-operated temperature control valve on the bypass line and an air-operated temperature control valve at the heat exchanger discharge header, and both are controlled by the DCIS. These valves regulate the supply temperature. The bypass and discharge valve have the ability to be manually controlled. The bypass valve fails closed and the discharge valve fails open.
2. The PCW surge tank makeup flow is controlled by an air-operated block valve. The valve automatically opens and closes and can be manually controlled. The block valve opens when the PCW surge tank level drops to a predetermined low level. The block valve closes when the surge tank level rises to a predetermined high level. The surge tank makeup water inlet block valve fails closed.
3. The cooling water temperature to the Turbine Lube Oil coolers is regulated by two temperature control valves, one at the Turbine Lube Oil cooler discharge and one on the bypass around the cooler, which operate in split range control mode. This type of flow will naturally balance the flow around the coolers. The bypass and discharge valves have the ability to be manually controlled. The bypass valve fails closed and the discharge valve fails open.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

4. The cooling water temperature to the Generator Air Coolers is regulated by two temperature control valves, one at the Generator Air Coolers discharge and one bypass around the coolers, which operate in split range control mode. This type of control will naturally balance the flow around the coolers. The bypass and discharge valves have the ability to be controlled manually. The bypass valve fails closed and the discharge valve fails open.
5. The cooling water temperature to the Shutdown Cooling Heat Exchangers is regulated by two temperature control valves, one at the Shutdown Cooling Heat Exchanger discharge and one bypass around the cooler, which operate in split range control mode. This type of control will naturally balance the flow around the heat exchangers. The bypass and discharge valves have the ability to be controlled manually. The bypass valve fails closed and the discharge valve fails open.
6. Upon a low surge tank level signal, the standby train of PCW isolates the Reactor Component Cooling Water Piping Distribution line and automatically places any non-operating equipment in bypass. The train that continues to lose water will eventually trip on a low suction head signal, but at least one train of PCW is preserved in order to support nuclear defense in depth functions.
7. Normally one PCW pump and heat exchanger train is in service with the other on standby. A pump trip signal results in the starting of the standby PCW train and CWS train.
8. The PCW Heat Exchangers interfaces directly with the CWS system. PCW trains cannot be started until the associated CWS train is running. The system connects such that an automated standby train of either CWS or PCW results in automatic realignment. The isolation valves are automatic with manual overrides available.

PCW controls, displays, and alarms include, but are not limited to, the following:

Main Control Room Panel Controls:

- PCW pump(s) start/stop and pump selection controls
- PCW heat exchanger(s) outlet valve open/close controls
- PCW supply temperature control valve(s) controls
- Surge Tank Makeup water valve open/close control
- PCW Heat Exchanger temperature control Air Operated Valves (AOVs)

Main Control Room Displays:

- PCW pump discharge pressures
- PCW pump operation status
- PCW operating temperatures (All heat exchanger outlet temperatures)
- PCW surge tank levels
- AOV open/close status
- PCW system flow rates

Main Control Room Alarms:

- PCW pump header low/high discharge pressure

- PCW cooling water supply high outlet temperature
- PCW cooling water supply low outlet temperature
- PCW surge tank high and low levels
- PCW heat exchanger high differential pressure

#### **9A.2.1.8 Monitoring, Inspection, Testing, and Maintenance**

The PCW is designed such that major equipment is provided adequate equipment removal paths and personnel access points.

The equipment and components of the PCW are designed for inspection and maintenance during plant operation without requiring complete loss of the Turbine Building cooling loop or the Reactor Building cooling loops.

Provisions to isolate the Turbine Building cooling loop or the Reactor Building cooling loops using AOV is provided. This design feature supports inspection, testing and maintenance activities.

Provisions for PCW drainage to the Equipment and Floor Drain System (EFS) are provided due to its high probability of having chemical content. If the PCW captured water is not chemically contaminated, the water is recycled back to the PCW surge tank.

Routine testing of the PCW system is conducted in accordance with normal power plant requirements for demonstrating system and component functionality and integrity. This includes testing for heat exchanger performance, surge tank levels and water quality standards.

The PCW design includes provisions to take periodic samples for analysis to ensure the water quality meets the chemistry specifications.

#### **9A.2.1.9 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are as low as reasonably achievable in operational states and in accident or post-accident conditions.

#### **9A.2.1.10 Performance and Safety Evaluation**

System reliability is enhanced through the use of periodic system testing, and preventive maintenance.

The PCW includes the components necessary to provide cooling water in support of the redundant Reactor Component Cooling Water Piping Distribution and the Turbine Component Cooling Water Piping Distribution. Portions of the PCW necessary to support Safety Class 3 functions to provide temperature control of the fuel pool, and cooling for the SDC heat exchanger, are designed to Non-Seismic Category, Quality Group D requirements. The redundant design elements of the PCW design ensures that the safety function of the system is maintained.

#### **9A.2.1.11 References**

- 9A.2.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.1-2 ANSI/HI 1.3, "Rotodynamic Centrifugal Pumps for Design and Application," Hydraulic Institute.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

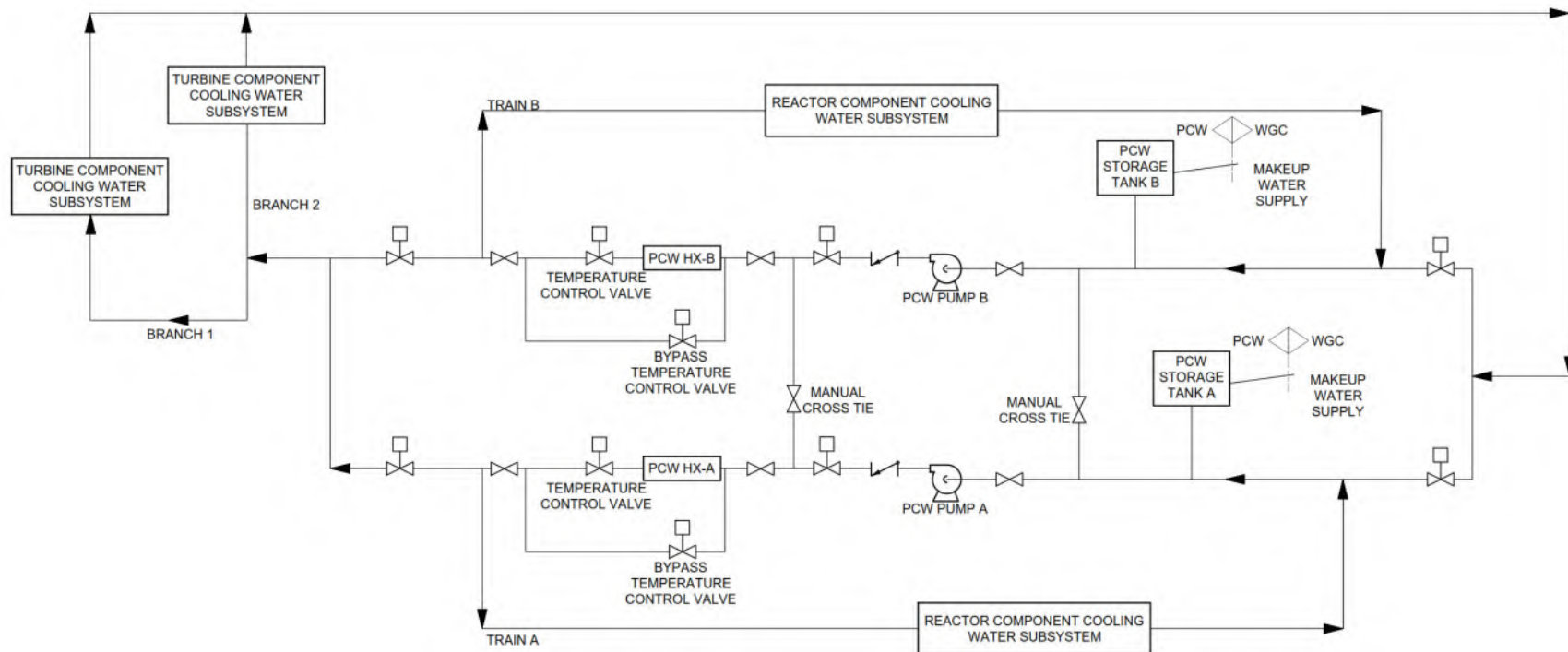
**Table 9A.2.1-1: Plant Cooling Water System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Instrumentation and Control System	Provides SC3 Instrumentation and Control. Pump logics, valve logics, instrumentation (pressure, flow, temperature, level) etc.	Safety Class 3 Instruments
Non-Safety Instrumentation and Control System	Provides Non-Safety Class (SCN) Instrumentation and Control Pump logics, valve logics, process variables, etc.	(SCN) Instruments
Process, Radiation and Environmental Monitoring System	Provides sampling points for chemical analysis	PCW Pump Common header
ICS Pool Cooling and Cleanup System	Provides cooling water to ICS pool cooler to ICS pool cooler from Reactor Component Cooling Water Piping Distribution	ICS Pool Cooler
Shutdown Cooling System	Provides cooling water to SDC cooler from Reactor Component Cooling Water Piping Distribution	SDC Cooler
Fuel Pool Cooling and Cleanup System	Provides cooling water to FPC cooler from Reactor Component Cooling Water Piping Distribution	FPC Cooler
Condensate and Feedwater Heating System (CFS)	Provides cooling water to Condensate and Feedwater Pumps and their Variable Frequency Drives from Turbine Component Cooling Water Piping Distribution	Condensate and Feedwater Variable Frequency Drive and Motors
Main Turbine Equipment (MTE)	Provides cooling water to Lube Oil and Electro-Hydraulic Cooler (EHC) from Turbine Component Cooling Water Piping Distribution	Lube Oil and EHC Coolers
Generator, Exciter, and Isophase Bus Ducts	Provides cooling water to the Generator cooler and Isophase Cooler from the Turbine Component Cooling Water Piping Distribution	Generator and Isophase Coolers
Main Condenser and Auxiliaries	Provides cooling water to the Vacuum Pumps skids from Turbine Component Cooling Water Piping Distribution water loop	Vacuum Pump Skids
Circulating Water System	Circulating Water acts as a heat sink to PCW heat exchangers	PCW Heat Exchangers

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Plant Pneumatic System	Provides cooling water to the Plant Pneumatics System from Reactor Component Cooling Water Piping Distribution PPS Supplies instrument air to PCW Valves	PPS coolers  Valve actuators
Safety Class 2 and 3 Electrical Distribution System	Provides PCW components power	SC3 pumps and valve motors
Non-Safety Electrical Distribution System	Provides Non-Safety PCW components power	SCN motors
Equipment and Floor Drain System	Provides drainage and collection for contaminated or potentially contaminated waste	Drain hub
Water, Gas, and Chemical Pads	Water, Gas and Chemical Pads provides demineralized makeup water to the PCW surge tanks	Surge Tank

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 9A.2.1-1: Plant Cooling Water**

## **9A.2.2 Reactor Water Cleanup System**

The Reactor Water Cleanup System (CUW) provides blowdown-type cleanup flow for the Reactor Pressure Vessel (RPV) during the reactor power operating mode. Cleanup or filtration and ion removal is performed by the CFD (Chapter 10, Subsection 10.3.1). The CUW provides an overboarding flow path to the condenser hotwell (condensate pump suction) or Liquid Waste Management System (LWM) (Chapter 11, Section 11.2) directly from the RPV lower region. CUW piping can be utilized to reduce reactor temperature stratification with reverse flow from the Shutdown Cooling System (Subsection 9A.2.3).

### **9A.2.2.1 System and Equipment Functions**

Refer to Chapter 1, Section 1.8 for a description of plant operating modes.

The CUW performs the following functions during normal and off-normal conditions.

#### **9A.2.2.1.1 Normal Functions (Non-Safety-Category)**

- Mode A
  - Normal Reactor Water Cleanup (1% Flow) Plant mode 1, 2, 3, 4)
- Mode B – Overboarding
  - Mode B1-A – Overboarding from Nuclear Boiler System (NBS)(Chapter 5) to Main Condenser and Auxiliaries (MCA) (Chapter 10, Section 10.5) (Plant mode 1, 2, 3, 4)
  - Mode B1-B – Overboarding from NBS to Liquid Waste Management System (LWM) (Chapter 11, Section 11.2) (Plant mode 1, 2, 3, 4)
  - Mode B2-A – Overboarding from SDC through CUW to MCA (Plant mode 2, 3, 4, 5)
  - Mode B2-B – Overboarding from SDC through CUW to LWM (Plant mode 2, 3, 4, 5)
- Mode C
  - RPV thermal stratification reduction (plant mode 2)

### **Overboarding**

During overboarding, the heat exchanger is in service to cool the reactor water to minimize flashing and two-phase flow in the pressure reducing components and downstream piping. Flow can be directed to the condenser hotwell, or in the event high radiation is detected, the overboarding flow is manually shifted to the LWM.

### **Thermal Stratification Reduction**

The system is designed to provide flow through the bottom head connections in the CUW during startup operations to reduce thermal stratification caused by the continuous input of cold Control Rod Drive (CRD) flow through the control rod drives.

### **Cleanup Flow to Condenser**

During Plant Operational Modes 1, 2, 3, and 4, CUW provides the equivalent of one percent of Feedwater (FW) nominal flow to downstream of a condensate pump for remixing and filtration.

#### **9A.2.2.1.2 Normal Function (Safety-Category)**

To ensure the safety of the plant, the system continuously monitors for leakage utilizing density compensated differential flow measurements. If a leak is detected, the containment isolation

function isolates the system from the Reactor Coolant Pressure Boundary (RCPB) through the closure of the containment isolation valve.

#### **9A.2.2.1.3 Off-Normal Functions (Non-Safety Category)**

The CUW system does not perform any Non-Safety-Category functions during off-normal conditions.

#### **9A.2.2.1.4 Off-Normal Functions (Safety Category)**

The CUW system performs Leak Detection and Isolation functions in Off-Normal conditions. Off-Normal functions for leak detection are the same as described in Subsection 9A.2.2.1.3.

Containment isolation is provided in accordance with REGDOC 2.5.2, Section 8.6.6 (Reference 9A.2.2-1). Leak detection is provided in accordance with Chapter 6, Subsection 6.3.2.8. Chapter 6, Subsection 6.3.4 describes the BWRX-300 containment isolation and containment isolation valves.

#### **9A.2.2.2 Safety Design Bases**

The following represents the CUW safety design bases:

1. DL3 leak detection actuates CUW isolation on CUW line break indication in Modes 1, 2, 3, and 4.
2. As part of DL3, the CUW provides a containment isolation valve (Safety Class 1) on piping that penetrates the containment boundary. The containment isolation valves are designed to close upon receiving an isolation signal from Safety Class 1 Instrumentation and Control System.
3. As part of DL3, upon detecting a CUW line break in Modes 1, 2, 3, and 4 the Safety Class 1 Instrumentation and Control System actuates CUW isolation. The CUW isolation valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
4. Defense Line 4a (DL4a) leak detection actuates CUW line isolation on break indication in the CUW lines in Modes 1, 2, 3, and 4. Upon detecting a CUW line break in modes 1, 2, 3, and 4 the Safety Class 2 and 3 Instrumentation and Control System actuates CUW isolation. The CUW isolation valves are designed to close upon receiving an isolation signal from Safety Class 2 and 3 Instrumentation and Control System.

#### **9A.2.2.3 Description**

Figure 9A.2.2-1 depicts a simplified flow diagram of the CUW and interfacing systems.

The CUW system consists of one train. The CUW receives flow through two nozzles located on the RPV. The train's inlet is independently connected to RPV penetrations located at about the mid-vessel height which take inlet flow from nozzles located near the RPV bottom head. This piping up to the RPV isolation valves are a part of the NBS. The inlet piping connects to the reactor vessel and combines inside containment to form one discharge line. This line is provided with a containment isolation valve where it penetrates containment. This valve will receive a signal from the CUW leak detection system and will close upon a detected leak. CUW continues through a regenerative heat exchanger and a pressure reduction station. The heat exchanger and pressure reduction station are designed to condition the water to acceptable temperatures and pressures for processing to the condensate system or overboarding. The heat exchanger is designed to recover heat back to the condensate and feedwater system. Discharge piping is



connected either to a condensate line for the normal CUW function or routed to the condenser hotwell or LWM for overboarding.

The leak detection and isolation subsystem perform both a DL3 and DL4a function to identify leaks and line breaks in the CUW and to initiate automatic response actions by other systems. For DL3, in MODES 1-4, the detection of a leak or line break in the CUW initiates isolation of the CUW system. For DL4a, in MODES 1-4, the detection of a leak or line break in the CUW initiates isolation of the CUW system.

The leak detection and isolation subsystem utilize a single Flow Element (FE) in the intake flow path and a single FE in each discharge flow path. Each FE serves independent, redundant sets of divisional Flow Transmitters (FTs), three FTs each for the DL3 and the DL4a line break detection functions. The basis for this arrangement is that a leak or break near the end of a discharge flow path may not be detectable by a single FE/FT feature located in the intake location of the system due to the physical distance between those locations. The design is intended to ensure that a leak or break can be identified by any of the FE/FTs at a detection point, or by differential flow between an upstream and downstream FE/FT detection point. Hence the use of detection features at the intake and at the discharge flow paths allows for detection of differential flow between the intake and any discharge flow path, or detection of a leak or break within detection proximity of any single FE.

The FT readings are density compensated by sets of three independent, redundant SC1 Temperature Transmitters. This feature will account for density changes in the water across the heat exchanger that is between the FE located in the intake flow path and the FEs located in each discharge flow path. Density compensation is used to avoid spurious isolation of the CUW and spurious generation of a signal that would initiate unwanted response actions by other systems.

Environmental Qualification of Safety-Class SSC's is provided as applicable as discussed in Chapter 3, Section 3.9.

#### **9A.2.2.3.1 Component Description**

The following provides description of CUW components.

##### **Piping and Valves**

The system piping is sized to ensure it can withstand the maximum pressure and temperature combination while keeping the maximum flow velocities within the acceptable limits.

The piping and the wetted parts of the valves are stainless steel to help with ALARA (Chapter 12, Section 12.3) concerns since this system is in contact with potentially radioactive water. Smooth bends are used instead of welded fittings where practical. At certain locations within the system, stainless steel pipes with electro-polished finish are used to reduce corrosion product buildup.

All valves in the system are used for isolation (on/off) except for a throttling valve in the pressure reduction station. This valve is used to throttle the system to achieve desired flow rates.

The CIVs for CUW fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.

- Overpressure protection is provided per the applicable design code, as follows:
  - ASME Boiler and Pressure Vessel Code, Section III for Class 1, 2, and 3 piping and components
  - ASME B31.1, Power Piping for Non-Safety-Category function piping
  - ASME BPVC, Section VIII for pressure vessels

Radioactive equipment and piping are designed to minimize crud buildup and to provide for easy decontamination and maintenance.

### **Regenerative Heat Exchanger**

The CUW heat exchanger is a regenerative heat exchanger cooled by the Condensate and Feedwater Heating System flow (Chapter 10, Subsection 10.3.2). The heat exchanger is designed to reduce the temperature of the inlet flow. Most of the heat of the CUW water is transferred to the condensate returning to the reactor, thus losing almost no thermal value.

The CUW heat exchanger is designed, manufactured, installed, and tested in accordance with applicable codes and standards, including Thermal Exchanger Manufacturers Association, API-662, ASME BPVC, Section VIII, Division 1. The heat exchanger tubes are seamless and constructed of stainless steel or titanium to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger is designed with provisions for tube replacement as well as individual tube plugging. An excess of tubes factored into the design of the heat exchanger accommodates tube plugging.

### **System Pressure Reduction Station**

At the pressure reduction station, a breakdown orifice (or pressure reduction valve) is used to reduce pressure before the water enters the CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD cleanup filters and demineralizers. This pressure is greater than condensate pressure at the connection point to ensure positive flow. The breakdown orifice (or pressure reduction valve) and associated instrumentation are located downstream of the heat exchanger and upstream of the condensate pump discharge tie-in location in the TB.

#### **9A.2.2.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC (as applicable) through material chemistry, heat treatment, contamination and material processes controls.

#### **9A.2.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.2-1 which provides the description and boundary for each interfacing system.

#### **9A.2.2.6 System and Equipment Operation**

##### **9A.2.2.6.1 Initial Configuration (Pre-Startup)**

The CUW is initially idle with the system isolation valves closed. The CUW is shutdown water solid, but because it is idle for long periods, the system pressure may need to be checked to ensure there was no leakage while shutdown. If necessary, the system piping is filled and vented to ensure there is no water hammer upon CUW startup.

##### **9A.2.2.6.2 System Startup**

When the CUW is actuated from the main control room, it first enters a stage of pre-warming. The system SSC are at ambient temperature due to inactivity in Plant Operating Mode 1. The water coming from the RPV is expected to be much warmer; therefore, this pre-warming stage is important to allow the system to gradually heat up to prevent over stressing the components due to thermal growth. The pre-warming is controlled by slowly initiating flow through the CUW through the use of the flow control valve and eventually ramping up to full flow before transitioning to normal operations.

#### **9A.2.2.6.3 Normal Operations**

Operations under normal conditions are described below. Table 9A.2.2-2 presents the plant operating Modes and corresponding CUW operational modes.

Mode A – Power Operation:

During power operation (plant Mode 1), the CUW system is in-service, taking input from the bottom of the RPV through two inlet lines. Water flows through piping internally mounted inside the RPV downcomer region and exits the vessel about four metres above top of active fuel. At the top of the piping outside the RPV, there are double isolation valves integral to the vessel which close to prevent RPV inventory from exiting the CUW line in case of a pipe break. Except for the containment isolation and leak detection related functions, the CUW is an SCN system, and all other system functions defined below are SCN.

The two inlet lines join into a single line outside the RPV where it exits the containment vessel, routes through piping in the steam tunnel and into the Turbine Building. The system contains a regenerative heat exchanger in which the CUW water is cooled by the Condensate and Feedwater Heating System. Most of the heat of the CUW water is transferred to the condensate returning to the reactor. A breakdown orifice is used to reduce pressure before the water enters CFS downstream of the condensate pumps where it is mixed with the condensate and cleaned in the CFD (Chapter 10, Subsection 10.3.1) cleanup filters and demineralizers.

During plant mode 2, 3, & 4 CUW can operate in Mode A. However, due to relying on the reactor for motive pressure for flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow being maintained through throttling as reactor pressure increases.

Mode B1-A:

When there is sufficient pressure in the RPV for flow in CUW, CUW can overboard to the condenser hotwell. In this mode CUW operates the same as Mode A, with the exception of flow being diverted to the condenser hotwell. This function is available in plant modes 1, 2, 3, & 4.

Mode B1-B:

When there is sufficient pressure in the RPV for flow in CUW, CUW can overboard to LWM. In this mode CUW operates the same as mode A, except for flow being diverted to LWM. This function is available in plant Modes 1, 2, 3, & 4.

Mode B2-A:

When there is insufficient pressure in the RPV for flow in CUW, SDC can overboard through the CUW to the condenser hotwell. SDC interfaces with CUW in the steam tunnel, and continues on the same path as described in mode B1-A. Mode B2-A is available in plant modes 2, 3, 4, & 5.

Mode B2-B:

When there is insufficient pressure in the RPV for flow in CUW, the SDC can overboard through the CUW to LWM. SDC interfaces with CUW in the steam tunnel, and continues on the same path as described in mode B1-B. Mode B2-B is available in plant modes 2, 3, 4, & 5.

Mode C:

To help reduce vessel stratification, caused by continuous input of cold CRD flow through the control rod drives, while shut down and during preparation for startup, the SDC water can be routed to the RPV lower region through the normal CUW inlet lines. The SDC thermal stratification

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

reduction line interfaces with the CUW system between the regenerative heat exchanger and the isolation containment valve. Mode C is only available in plant mode 2.

**Mode 1 – Power Operation:**

During power operation, the CUW system is in service in system Mode A, with flow from the bottom of the RPV through two inlet lines. Water flows through piping internally mounted inside the RPV downcomer region and exits the vessel about four metres above top of active fuel. At the top of this piping outside the RPV, there are double isolation valves integral to the vessel, which prevents RPV inventory from exiting the CUW line in case of a pipe break.

The two inlet lines join into a single line outside the RPV, and the CUW water then is routed through piping in the steam tunnel. There is a regenerative heat exchanger in which the CUW water is cooled by the CFS. Most of the heat of the CUW water is transferred to the condensate returning to the reactor, thus losing almost no thermal value. A breakdown orifice is used to reduce pressure before the water enters CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD cleanup filters and demineralizers

**Mode 2 – Startup:**

During startup Mode, CUW operation is the same as during power operation. However, due to relying on reactor pressure for flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow increasing as reactor pressure increases.

During startup or shutdown, CUW can function to let down (overboard) the excess reactor inventory if there is enough pressure (CUW Mode B1-A and B1-B). If not, the SDC system can be used to let down the excess reactor inventory (CUW Mode B2-A and B2-B). The SDC letdown line interfaces with the CUW system upstream of the regenerative heat exchanger. Overboard flow can be directed to the condenser hotwell or LWM. In this mode, reactor vessel water level is maintained by the control rod drive water.

To help reduce vessel stratification during preparation for startup, SDC water can be routed to the RPV lower region through the normal CUW inlet lines (CUW Mode C).

**Mode 3 – Hot Shutdown:**

During hot shutdown operation, the CUW system is in-service if the RPV is pressurized. This mode is with reactor temperature  $>215.6\text{ }^{\circ}\text{C}$ . Vessel letdown can be performed as discussed above under the startup mode.

**Mode 4 – Stable Shutdown:**

During stable shutdown operation, the CUW system is isolated from the reactor. This mode is with reactor temperature  $\leq 215.6\text{ }^{\circ}\text{C}$  and  $>93.3\text{ }^{\circ}\text{C}$ . If CUW is isolated at the vessel, SDC can be used via CUW if overboarding is required.

**Mode 5 – Cold Shutdown:**

During cold shutdown operation, the CUW system is isolated at the vessel. This mode is with reactor temperature  $\leq 93.3\text{ }^{\circ}\text{C}$ . SDC can be used via CUW if overboarding is required.

**Mode 6 – Refueling:**

During refueling mode, the CUW is not in service.

#### **9A.2.2.6.4 Off-Normal Operations**

No manual operator actions are required to initiate or assure the completion of fundamental safety functions. The RPV isolation valves and containment isolation valve fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.

For large line breaks, it is noted that for FW line breaks and reactor water cleanup breaks outside of containment, leak detection systems identify the break condition and provide signals to close the affected system isolation valves.

#### **9A.2.2.6.5 System Shutdown**

The system is shut down in plant operating Mode 6.

Upon shutdown, the system isolation valves are closed. The system remains water solid while shutdown.

#### **9A.2.2.7 Instrumentation and Control**

During plant operation and after plant shutdown (with sufficient reactor pressure), CUW removes water from the bottom of the reactor and routes it to the condensate system or overboards excess "swell", when required, to the condenser hotwell or liquid radwaste. These functions require both monitoring and control.

The input signals to the control functions are triply redundant and the actuators receive two out of three voted commands from the CUW controller. The CUW controller is triply redundant and is dual ported to the plant balance of plant segment network. As with all Triple Modular Redundant (TMR) controllers, extensive hardware and software diagnostics are provided for operator monitoring and alarms. The CUW controller is also designed to support plant automation. Refer to Chapter 7 for information pertaining to TMR.

The CUW can be controlled from the main control room. The CUW controller can receive commands from the Plant Automation System (PAS) or be manually operated. The controller automates the operation of the system.

##### **9A.2.2.7.1 Flow**

There is one SC1 flow element along with triply redundant (SC1) transmitters on the system supply line at the containment boundary inside the steam tunnel. This flow indication is used in conjunction with identical flow elements and transmitters located at the end of each flow path. When used as a pair they monitor for system leakage. Main Steam Reactor Isolation Valves (MSRIV) and CUW Isolation are actuated on CUW line break indication in Plant Modes 1-4.

There are triply redundant SC2 transmitters independent from the SC1 flow element on the system supply line at the containment boundary. This flow indication is used in conjunction with identical flow elements and transmitters located at the end of each flow path. When used as a pair they monitor for system leakage. MSRIV Isolation and CUW Isolation are actuated on CUW line break indication in Plant Modes 1-4.

There are triply redundant SCN transmitters used for the control of the pressure reduction station.

##### **9A.2.2.7.2 Temperature**

There are triply redundant SC1 temperature sensors that accompany the SC 1 flow transmitters on the containment boundary and at the end of each flow path. These temperatures are used to provide input for the density compensation of the flow transmitters.

There are triply redundant Safety Class 2 (SC2) temperature sensors that accompany the SC1 flow transmitters on the containment boundary and at the end of each flow path.

There is one SCN temperature sensor downstream of the heat exchanger. This temperature is measured prior to entrance of the pressure reduction station to ensure that the temperature of the fluid is acceptable after the heat exchanger.

#### **9A.2.2.7.3 Pressure**

Pressure Reduction Station Inlet:

One set of triply redundant SCN pressure instruments are provided at the inlet of the pressure reduction station. These pressure instruments are used to control the pressure reduction station.

Pressure Reduction Station Outlet:

Three SCN pressure instruments are provided at the discharge of the pressure reduction station. These pressure instruments are used to monitor the discharge pressure of the pressure reduction station, to monitor status and performance.

#### **9A.2.2.7.4 Radiation**

CUW has a radiation monitor downstream of the heat exchanger provided by the PREMS. This monitor provides input to the Process and Radiation Monitoring Subsystem.

#### **9A.2.2.8 Monitoring, Inspection, Testing and Maintenance**

Maintenance and testing support equipment reliability. SSCs are designed to facilitate operation and maintenance. Maintenance activities, including post-maintenance and post-modification testing, are controlled by plant procedures. Maintenance activities restore SSCs to their original condition or involve a temporary alteration in accordance with plant procedures.

Maintenance practices consider industry best practices such as Institute of Nuclear Power Operations (INPO, etc.) and operating experience, and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities comply with Technical Specifications as required. Maintenance activities implement plant ALARA practices to minimize work activity dose. Maintenance activities involving critical plant equipment may require involvement of vendors or industry specialists.

The CUW is provided with chemical cleaning and decontamination connections that can utilize condensate to flush piping and equipment prior to maintenance to provide decontamination of the regenerative heat exchanger, pressure reduction station, and associated valves.

Inspections, checks, and tests conducted during the performance of maintenance activities are considered "In-Process Maintenance Tests" and are governed by the in-use maintenance procedures and processes.

Inspections, checks, and tests conducted following the conclusion of maintenance activities are considered "Post-Maintenance Tests" and are governed by plant Post-Maintenance Test procedures. ISI and IST requirements are established and include inspection/test frequency for SSCs. Remote monitoring of key parameters assists in the trending of component performance as part of condition-based predictive maintenance.

#### **9A.2.2.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. To control and

minimize personnel exposure, radioactive equipment is designed to minimize crud buildup and provide for decontamination and maintenance.

#### **9A.2.2.10 Performance and Safety Evaluation**

System reliability is enhanced through the use of periodic system testing, preventive maintenance, and application of redundant instrumentation.

The CUW SC1 functions during normal conditions and off-normal conditions include continuous monitoring for leakage utilizing density compensated differential flow measurements. If a leak is detected, in either normal or off-normal conditions the system is isolated from the reactor coolant pressure boundary using system isolation and containment isolation valves.

The arrangement of SSCs minimize the possibility of compromising the functionality of both the SC1 functions and their backup systems and equipment for any credible events that could cause damage to one region of the plant. The portions of the CUW responsible for performing containment isolation are located in the Reactor Building which provides protection against natural phenomena ensuring the ability of the CUW to perform its Safety-Category functions.

#### **9A.2.2.11 References**

- 9A.2.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.2.2-1: Reactor Water Cleanup System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	Locations where flow from the Reactor Pressure Vessel enter the CUW. The RPV isolation valves for reactor water cleanup will fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means	Physical boundary is at the dual NBS reactor isolation valves on two penetrations exiting the RPV
Shutdown Cooling System	Connection from SDC to CUW for overboarding flow at low pressures or flow to the vessel to reduce temperature stratification	The system interface is at the isolation valve between SDC and CUW
Condensate and Feedwater Heating System	Normal CUW flow to Condensate System. All cleanup of CUW injection flow is performed by CFD filters and demineralizers.  Coolant flow for CUW regenerative heat exchanger from Feedwater	Downstream of a Condensate pump prior to filters and demineralizers  CUW regenerative heat exchanger
Plant Pneumatics System	Provide air or nitrogen to operate air operated valves	The system interface is located at each valve actuator
Safety Class 2 and 3 Electrical Distribution System	Provide power to CUW instrumentation for system operation and line break detection	The system interface is at each powered component
Safety Class 1 Instrumentation and Control System	CUW instrumentation for leak detection provide input to Safety Class 1 Instrumentation and Control System. Safety Class 1 Instrumentation and Control System is the safety class instrumentation system	
Safety Class 2 and 3 Instrumentation and Control System	CUW instrumentation for system operation and leak detection provide input to Safety Class 2 and 3 Instrumentation and Control System	
Liquid Waste Management System	Overboarding to LWM to reduce vessel level if required and activity (radiation level) in the process flow may be high	The system boundary is after the overboard isolation valve



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Main Condenser and Auxiliaries	Overboarding to MCA to reduce vessel level if required and process flow activity (radiation level) is normal	The system boundary is after the overboard isolation valve
Process Radiation and Environmental Monitoring System	Monitor radiation levels of cleanup flow  Process sampling for radiological and chemistry control	Radiation monitor/transmitter on process flow line PREMS isolation valve

**Table 9A.2.2-2: Reactor Water Cleanup System Operating Modes**

<b>Mode</b>	<b>Title</b>	<b>Reactor Mode Switch Position</b>	<b>CUW System Modes</b>
1	POWER OPERATION (10% - 100% Reactor Power)	RUN	Mode A, B1-A, and B1-B
2	STARTUP	REFUEL <sup>(1)</sup> or STARTUP	All Modes
3	HOT SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Mode B1-A, B1-B, B2-A, and B2-B
4	STABLE SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Mode B1-A, B1-B, B2-A, and B2-B
5	COLD SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Modes B2-A and B2-B
6	REFUELING <sup>(2)</sup>	SHUTDOWN or REFUEL	NA

(1) All reactor pressure vessel head closure bolts fully tensioned

(2) One or more reactor pressure vessel head closure bolts less than fully tensioned

The diagram illustrates the Reactor Pressure Vessel (RPV) and its associated piping and systems, divided into 'INSIDE REACTOR CONTAINMENT' and 'OUTSIDE REACTOR CONTAINMENT'.

**INSIDE REACTOR CONTAINMENT:**

- REACTOR PRESSURE VESSEL:** The central component, shown as a vertical cylinder with a domed top.
- WARM-UP ISOLATION:** A section of piping containing an **AOV** (Automatic Overpressure Valve) and an **FE** (Flow Element).
- CLEANUP ISOLATION:** A section of piping containing a valve and an **FE**.
- REGENERATIVE HEAT EXCHANGER (RHXR-1):** A circular component with a central coil, connected to two **CFS** (Coolant Flow System) lines.
- PRESSURE REDUCTION STATION:** A section of piping containing an **FCV** (Flow Control Valve), an **RO** (Relief Valve), and a valve.
- LOW PRESSURE OVERBOARD:** A section of piping containing a valve and an **FE**.

**OUTSIDE REACTOR CONTAINMENT:**

- CONDENSER HOTWELL (MCA):** A component connected to the **LOW PRESSURE OVERBOARD** section via a valve and **FE**.
- LWM (Low Water Monitor):** A component connected to the **LOW PRESSURE OVERBOARD** section via a valve and **FE**.
- CUW (Cooling Water) and FW (Feed Water) connections:** Various connections to **CUW** and **FW** systems, including a **MCA** (Master Control Assembly) connection to the **CONDENSER HOTWELL**.

9A-52

### **9A.2.3 Shutdown Cooling System**

The SDC provides for decay heat removal when shutting down the plant. The system is also used to reduce reactor pressure vessel inventory and can be used in conjunction with CUW (Subsection 9A.2.2) piping to reduce reactor pressure vessel thermal stratification.

#### **9A.2.3.1 System and Equipment Functions**

##### **9A.2.3.1.1 Normal Functions (Non-Safety-Category)**

###### **Overboarding**

During reactor startup, it is necessary to remove the CRD purge water injected into the RPV as well as the excess reactor water volume arising from thermal expansion due to reactor heat-up. The SDC accomplishes these volume removals and thereby maintains proper reactor level until the reactor pressure is sufficient to allow use of the CUW overboard flow path.

During overboarding, the heat exchanger is in service to cool the reactor water to minimize flashing and two-phase flow in the downstream piping. The preferred overboarding destination is the hotwell that is part of the CFS (Chapter 10, Subsection 10.3.2); however, in the event high radiation is detected, the overboarding flow is manually shifted to the LWM (Chapter 11, Subsection 11.3).

Overboarding is considered part of SDC Mode B1 and is available in plant operating Modes 2-6.

###### **RPV Thermal Stratification Reduction**

The SDC is designed to provide flow through the CUW inlet lines into the bottom head region of the RPV during startup operations to reduce thermal stratification caused by the continuous input of cold CRD flow through the control rod drives.

Thermal stratification reduction is considered part of SDC Mode B2 and is available in plant operating Mode 2.

##### **9A.2.3.1.2 Normal Functions (Safety-Category)**

The SDC system for the BWRX-300 has two Safety Class functions:

###### **Decay Heat Removal (Defense Line 2/Safety Class 3)**

In conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the SDC is used to reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below saturation temperature at atmospheric pressure in less than one day.

The shutdown cooling function of the SDC provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures. The redundant trains of SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train.

Decay heat removal is considered part of SDC Mode A1 and is available in plant operating Modes 2-6.

###### **Leak Detection and Isolation (Defense Line 3/Safety Class 1 and Defense Line 4a/Safety Class 2)**

When the SDC is operational, the system continuously monitors for leakage utilizing differential flow measurements. If a leak is detected, the system is isolated from the reactor through a

number of system isolation and containment isolation valves to prevent the excessive reduction of reactor water inventory.

#### **9A.2.3.1.3 Off-Normal Functions (Non-Safety Category)**

If feedwater flow is lost in operating Modes 2-4, the operator initiates either the SDC or ICS (Chapter 5, Section 5.8) to reduce reactor water temperature and pressure.

Upon identification of a small line break inside containment with or without preferred power during operating Modes 3-4, the SDC is initially used for decay heat removal until the containment is isolated at which time the ICS is initiated to remove the decay heat.

#### **9A.2.3.1.4 Off-Normal Functions (Safety-Category)**

The system does not perform any safety category functions during off-normal conditions.

The design of the SDC system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.2.3-1) as related to the design of containment isolation valves. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

#### **9A.2.3.2 Safety Design Bases**

The SDC has two Safety-Class functions, Decay Heat Removal, and Leak Detection and Isolation. Refer to Subsection 9A.2.3.1.2 for discussion of SDC Safety-Category functions.

Plant systems retain sufficient heat removal capacity to prevent boiling in the reactor during periods of plant shutdown with the Reactor Pressure Vessel head removed, and to simultaneously prevent boiling in the fuel pool, assuming a Loss of Preferred Power event and the most limiting single active failure.

Defense lines associated with Leak Detection and Isolation are noted below.

1. As part of Defense Line 3 (DL3) 3, the Shutdown Cooling isolation valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
2. As part of DL4a, the SDC isolation valves are designed to close upon receiving an isolation signal.
3. As part of DL2, the SDC is designed to initiate upon receiving an initiation signal from Safety Class 2 and 3 Instrumentation and Control System.
4. DL2 operators are able to manually initiate SDC train in Modes 2-6.
5. DL3 leak detection actuates MSRV Isolation on FW and SDC Line break indication in Modes 1, 2, 3, and 4.
6. DL3 leak detection actuate FW and SDC line isolation on FW and SDC line break indication in Modes 1, 2, 3, and 4.
7. DL4a Leak Detection actuates MSRV Isolation on FW line break indication / SDC Line break indication in Mode 1-4.
8. DL4a Leak Detection actuates FW Isolation/SDC line isolation on FW line break indication/SDC Line break indication in Mode 1-4.

#### **9A.2.3.3 Description**

The SDC comprises two independent pump and heat exchanger trains. These trains together provide redundant decay heat removal capacity such that each train is designed to remove 100%

of decay heat as soon as 4 hours after reactor shutdown. The major components of each train are a pump and a Heat Exchanger (HX), along with valves, piping, instrumentation and controls, and power inputs. The two trains operating in parallel provide the systems full rated shutdown cooling performance. Bypass lines and valves are included around the tube side of each HX to allow bypassing of the HX for SDC functions such as reducing RPV thermal stratification.

Each train's suction is independently connected to a separate ICS (Chapter 5, Section 5.6) condensate return line outside of containment, and downstream of the ICS containment isolation valves. Each SDC train's return piping is independently connected to a separate CFS (Chapter 10, Subsection 10.3.2) feedwater line outside of the containment isolation valves. This arrangement helps to prevent short circuiting of the flow inside of the RPV since the SDC suction is taken from the ICS return nozzle which originates in the chimney region over the core and the SDC return flow utilizes the CFS return which terminates outside this region in the downcomer area.

Fittings are registered in compliance with Technical Standards and Safety Authority.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSCs.

#### **9A.2.3.3.1 Decay Heat Removal Subsystem**

The subsystem comprises the flow path from the ICS to the SDC pump, through the tube side of the SDC heat exchanger, and then returned to the RPV through the CFS as depicted in Figure 9A.2.3-1. The subsystem is available in Plant Modes 2 through 6, though the condenser and ICS are expected to be used initially to cool the reactor after shutdown. The subsystem is initially operated at low flow to allow the system Structures, Systems, and Components, to be brought up to operating temperature without overstressing the components. Once the system is up to temperature, the flow rate increases and is controlled to maintain PCW (Subsection 9A.2.1) heat exchanger cooling water exit temperature within the interface requirements of 54.4 °C for two SDC trains running and 60.0 °C for one train running.

#### **9A.2.3.3.2 Overboarding Subsystem**

The overboarding subsystem supports RPV level control during Plant Modes 2 through 6 when the fuel pool gate is installed and the CUW (Subsection 9A.2.2) is not available for use. The subsystem comprises the flow path from the ICS to the SDC pump, through the tube side of the SDC heat exchanger, and then through the overboard flow path (see Figure 9A.2.3-2). The figure shows multiple overboard flow pathways highlighted; however, only one overboard path is implemented at a time by controlling the position of the valves on each flow path. The SDC trains share the overboard control valves and each train has its own isolation valve to prevent inadvertent overboarding. Each overboard flow path has been designed to provide adequate overboarding flow to support the worst-case scenario (fastest) heat-up rate of the reactor combined with the expected water input from the CRD (Chapter 4, Section 4.5).

The preferred overboard flow path is from the SDC to the CFS hotwell; however, in the event that high radiation is detected in the effluent, the flow path is manually switched from the CFS to the LWM system. The overboarding flow pathways through the CUW are utilized in the event of failure of any of the other overboard pathway isolation valves, or if the pressure in the SDC is too high for the interfacing system, in which case, the CUW pressure reduction station is utilized to reduce the fluid pressure to protect the interfacing system.

Each overboard flow pathway from the SDC has two fail-closed valves in series. There are two valves in series to prevent exposing a lower design pressure system or subsystem to a higher design pressure system in the event of an inadvertent opening of a single valve.

#### **9A.2.3.3.3 Thermal Stratification Reduction Subsystem**

The subsystem comprises the flow path from the ICS to the SDC pump, through the SDC heat exchanger bypass line, and then through the overboard flow path through the CUW and back to the RPV through the NBS as depicted in Figure 9A.2.3-3. The subsystem provides approximately 200 gpm to the bottom region of the RPV to counter act the constant 20 gpm flow from the CRD through the control rod drives. The 10:1 flow ratio is expected to provide sufficient heat to overcome the cooling effect of the CRD flow. Also, the velocity at the discharge of the piping into the RPV is sufficient to promote mixing in the bottom head region to minimize thermal stratification.

#### **9A.2.3.3.4 Leak Detection and Isolation Subsystem**

The leak detection and isolation subsystem are comprised of three flow elements per train along with two sets of triply redundant transmitters per flow element to support both DL3 and DL4a leak detection requirements. The readings are utilized to account for temperature change across the heat exchanger. This is done to avoid spurious isolation of the SDC which would reduce the SDC capacity, or if both trains are isolated, would require the restart of ICS. Restart of ICS is only possible if the reactor is still pressurized.

The control system compares the flow readings from the supply flow transmitters against flow readings from the return and overboard flow transmitters. If the system detects a flow differential greater than the acceptable leakage value, the control system actuates MSRIV Isolation, Feedwater Isolation, and SDC Isolation.

#### **9A.2.3.3.5 Component Description**

##### **Shutdown Cooling System Piping and Valves**

The SDC piping material is 304L stainless steel. The pipe wall thickness has been sized to ensure it can withstand the maximum pressure and temperature combination while the diameter has been chosen to ensure the maximum flow velocities stay within the acceptable limits.

The piping and the wetted parts of the valves are stainless steel which supports ensuring ALARA goals since this system is in contact with potentially radioactive water.

All valves in the system are used for isolation (on/off) except for the heat exchanger bypass valve. The bypass valve is used to throttle the bypass around the heat exchanger and can be used in conjunction with the SDC pump speed control to control the exit temperature of the heat exchanger.

SDC decay heat removal piping and valves are designed, manufactured, and tested in accordance with ASME BPVC Section III, Class 3. SDC overboard piping and valves are designed, manufactured, and tested in accordance with ASME B31.1.

##### **Shutdown Cooling System Pumps and Adjustable Speed Drives**

Each train of the SDC contains one SC3 horizontal centrifugal pump sized to allow sufficient flow to remove 100% of the decay heat generated by the core 4 hours after shutdown. The SDC pumps are designed to pump the maximum flow required by the system to meet SDC cooldown requirements for decay heat removal. The pump is installed below the RPV inlet nozzle which provides adequate NPSH at maximum flow conditions. Each pump is paired with an adjustable speed drive to allow for control of the cooldown rate and the reactor temperature during shutdown.

The adjustable speed drive is necessary to allow the flow to be adjusted to meet all expected system flow design points.

SDC pumps are designed, manufactured, and tested in accordance with ASME BPVC Section III, Class 3. SDC pump wetted parts are manufactured from stainless steel.

### **Shutdown Cooling System Heat Exchangers**

Each train of the system contains one SC3 shell and tube heat exchanger sized with sufficient capacity to remove 100% of the decay heat generated by the core 4 hours after shutdown. The heat is transferred from the SDC process fluid to the PCW.

SDC heat exchangers are designed, manufactured, installed, and tested in accordance with ASME BPVC Section VIII, Class 3. SDC heat exchanger tubes are constructed of stainless steel.

#### **9A.2.3.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC through material chemistry, heat treatment, contamination, and material processes controls.

#### **9A.2.3.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.3-1 for SDC interfaces with other equipment or systems.

#### **9A.2.3.6 System and Equipment Operation**

Plant cooldown following shutdown is accomplished using a combination of the main condenser, ICS, and SDC systems. Although the SDC system is qualified for high temperature and pressure operation, the main and ICS condensers are the preferred cooldown sources immediately following shutdown.

##### **9A.2.3.6.1 Initial Configuration (Pre-Startup)**

The system is initially idle with the system isolation valves closed. When the system is shutdown, it is water solid, but because it is idle for long periods, the system pressure may need to be checked to ensure there was no leakage while shutdown. If necessary, the system piping is filled and vented to ensure there is no water hammer upon system startup.

##### **9A.2.3.6.2 System Startup**

When the system is actuated from the control room, it first enters a stage of pre-warming. It is expected that the system SSCs are at ambient temperature due to inactivity in Plant Mode 1, but the water coming from the RPV is expected to be much warmer; therefore, this pre-warming stage is important to allow the system to gradually heat up to prevent over stressing the components due to thermal growth. The pre-warming is controlled by initiating low flow through the SDC using the adjustable speed drive and eventually ramping up to full flow before transitioning to normal operations.

##### **9A.2.3.6.3 Normal Operations**

The normal and off-normal operational modes of the SDC are described below. This includes how the plant transitions from a shutdown condition to full power, how steady-state full power operation is maintained, how the transition from full power to shutdown is achieved, and how a refueling outage is performed. Table 9A.2.3-2, "BWRX-300 Mode Table" lists the plant operational modes and the corresponding operational modes of SDC. The SDC system modes and functions are discussed in more detail below.

#### **Plant Mode 1 - Power Operation**

During power operation, the SDC is not in-service.



### **Plant Mode 2 - Startup Mode**

The SDC is a manually initiated system.

During startup operation, the SDC is initially in-service to provide a reactor water reject flow path for RPV level control. The SDC is eventually taken out of service when the RPV pressure is high enough to establish CUW flow and when the SDC has inadequate net positive suction head due to voiding in the chimney section.

If the SDC is required to reduce vessel temperature stratification, SDC Mode B2 is utilized. After flow from SDC enters the CUW it is routed reverse the normal flow direction and discharges from the CUW inlet lines into the RPV bottom head region.

### **Plant Mode 3 - Hot Shutdown Operation**

During hot shutdown operation, the SDC is not normally in-service since heat removal by steam condensation is more efficient. The SDC requires adequate water level in the chimney after shutdown before it is placed into service.

### **Plant Mode 4 - Stable Shutdown Operation**

During hot shutdown operation, the SDC is not normally in-service since heat removal by steam condensation is more efficient. The SDC requires adequate water level in the chimney after shutdown before it is placed into service.

When RPV pressure drops to a point that the normal reactor water reject flow path is ineffective, the SDC can be placed into service to provide this function through an interface connection to the CFS and LWM.

### **Plant Mode 5 - Cold Shutdown Operation**

During cold shutdown operation, the SDC is in-service. Each train of SDC can separately remove 100% of the decay heat generated 4 hours following reactor shutdown. Both trains of the SDC are normally available in cold shutdown conditions, but one train may be taken out of service at a time for maintenance if needed.

With the RPV depressurized, the SDC also provides water reject flow through an interface connection to the CFS and LWM.

### **Plant Mode 6 - Refueling Mode**

At any time when the fuel pool gate is installed and water level is being held constant, the SDC is overboarding to the LWM to offset the CRD purge flow.

During refueling operation, the SDC system is in service. Each train can separately remove 100% of the decay heat 4 hours following reactor shutdown. The SDC system reject flow path is coordinated with the water level in the RPV, position of the FP gate, and availability of the Fuel Pool Closed Cooling and Cleanup System. When fully flooded and connected to the fuel pool, the Fuel Pool Cooling and Cleanup System can maintain a constant level. Refer to table 9A.2.3-3 for information pertaining to SDC configuration during refueling mode.

#### **9A.2.3.6.4 Off-Normal Operation**

##### **Loss of Feedwater Flow Plant Modes 2-4**

Operator initiates SDC or ICS.

##### **Small Line Break Inside Containment Plant Modes 3-4**

On identification of a small line break inside containment with or without preferred power during Plant Modes 3-4, the SDC is used for decay heat removal until the containment is isolated at which time the ICS is initiated to remove the decay heat.

#### **9A.2.3.6.5 System Shutdown**

The SDC is shutdown as the plant heats up in Plant Mode 2. The cooling function is no longer required since heat is being intentionally added to the RPV and the overboarding function transitions from the SDC to the CUW when the RPV pressure is sufficient to drive the necessary flow through the CUW to the overboarding destination. The thermal stratification reduction function won't be necessary once the CUW is overboarding since any cold spots developing near the piping terminations are overboarded or mixed with the flow entering the core.

Upon shutdown, the pump is turned off and the system isolation valves closed. The system remains water solid while shutdown.

#### **9A.2.3.7 Instrumentation and Control**

The SDC is a manually initiated system but is designed for automatic control of functions such as SDC system piping pre-warming, adjusting the system heat removal rate to stay within vessel and HX cooling water limitations, as well as allowing the operator to input a temperature rate setpoint that allows automatic vessel water temperature cooldown rate. Additionally, during reactor heat-up, the SDC controllers receive flow demand signals from the RLC to allow "swell" to be overboarded to maintain reactor level. The SDC can be controlled from the Main Control room as well as the Secondary Control Room.

The SDC has its own (triply redundant) sensors for control and its own actuators. It is possible to validate the temperature signals used for control with reactor water temperature measurements and (when steaming) reactor pressure measurements. The SDC controllers are triply redundant and are dual ported to the plant nuclear segment network. As with all Triple Modular Redundant controllers, extensive hardware and software diagnostics are provided for operator monitoring and alarms. Refer to Chapter 7 for information pertaining to TMR.

##### **9A.2.3.7.1 Flow**

#### **Supply Flow**

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant Safety Class 2 transmitters on the system supply line upstream of each SDC pump. The instruments are used in conjunction with the return flow and overboard flow instruments to support the DL3 and DL4a function of monitoring for system leakage. MSRV Isolation, Feedwater Isolation, and SDC Isolation are actuated on SDC line break indication in Plant Modes 1-4.

The Safety Class 2 transmitters are also used to monitor the system flow rate to help the operator determine that the pump is functioning properly and as input into the control system for automatic control of the pump.

#### **Return Flow**

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant Safety Class 2 transmitters on the system return line downstream of the overboarding branch connection. This flow indication is also used in conjunction with the supply flow and overboard flow instruments to support the DL3 and DL4a function of monitoring for system leakage respectively.

#### **Overboard Flow**

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant SC2 transmitters on the system overboard line upstream of the point where the two system trains connect. This flow indication is also used in conjunction with the supply flow and return flow instruments to support the DL3 and DL4a function of monitoring for system leakage respectively.

#### **Pump Seal Purge Flow**

There is one Non-Safety Class flow element along with one Non-Safety Class flow transmitter on the pump seal purge line. This flow instrument is used to monitor flow from the CRD system to the SDC pump to ensure seal purge flow is within acceptable limits.

#### **9A.2.3.7.2 Temperature**

##### **Heat Exchanger Inlet Temperature**

There are triply redundant SC1 as well as independent triply redundant SC2 temperature sensors upstream of the pump. This temperature is used to provide input for the SC1 and SC2 supply flow transmitters respectively.

The SC2 temperature sensors along with the heat exchanger discharge temperature is used to monitor the performance of the SDC heat exchanger. In Plant Modes 3 and 4, when the SDC is operational, these temperature elements are used to determine the average reactor coolant temperature.

##### **Heat Exchanger Discharge Temperature**

There are triply redundant SC1 as well as independent triply redundant SC2 temperature sensors on the discharge of the heat exchanger. This temperature is used to provide input for the overboard flow and return flow transmitters.

The SC2 temperature sensors along with the heat exchanger inlet temperature is used to monitor the performance of the SDC heat exchanger. This temperature is also used to modulate the heat exchanger bypass valve to allow the combined discharge temperature from the heat exchanger and bypass to meet a pre-set value.

##### **Overboard Temperature**

There is one SCN temperature sensor on the overboard line. This temperature is measured prior to discharge to LWM, and CUW to ensure that the temperature of the fluid is acceptable for the interfacing system.

#### **9A.2.3.7.3 Pressure**

##### **SDC Pump Suction Pressure**

There is one SC3 pressure instrument provided at the suction of each SDC pump. This pressure instrument is used to monitor the suction pressure of the SDC pumps, in order to monitor pump status and performance.

##### **SDC Pump Discharge Pressure**

There is one SCN pressure instrument provided at the discharge of each SDC pump. This pressure instrument is used to monitor the discharge pressure of the SDC pumps, in order to monitor pump status and performance.

##### **Heat Exchanger Differential Pressure**

There is one SC3 differential pressure instrument provided to measure the pressure drop across the tube side of the heat exchanger. This pressure drop is monitored to help inform the operator of needed maintenance.

#### **9A.2.3.7.4 Radiation**

##### **Overboard Radiation Monitoring**

There is one SCN radiation monitor installed on the SDC overboard line to monitor radiation levels. Refer to Chapter 11, Section 11.5 for information pertaining to the functions of the instrument.

#### **9A.2.3.8 Monitoring, Inspection, Testing, and Maintenance**

Maintenance and testing support equipment reliability. SSCs are designed to facilitate operation and maintenance. ISI and IST requirements are established and include inspection/test frequency for SSCs. Remote monitoring of key parameters assists in the trending of component performance as part of condition-based predictive maintenance. Testing is performed to ensure required functional operability is maintained under design conditions. The Safety-Category Functions of the SDC that support decay heat removal are tested in accordance with ASME BPVC Section XI as applicable. Testing is performed in accordance with plant procedures. Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Section 14.3.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

#### **9A.2.3.9 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. In addition, the following design features are employed to reduce dose:

1. Process piping from the containment penetrations to the overboard lines should be routed through shielded areas or pipe chases to reduce the dose rates to the general areas of the reactor building.
2. Components within a compartment are arranged to facilitate rapid maintenance and inspection and allow for the addition of temporary shielding. Reducing the time for maintenance and inspection activities minimizes the duration and intensity of radiation exposure and assists with meeting ALARA goals.
3. Field located instruments and/or indicators are visibly and conveniently mounted outside of secondary shielding walls. This arrangement avoids unnecessarily exposing personnel to radiation during routine monitoring and maintenance.
4. SDC heat exchangers are manufactured with all-welded construction with flanged connections where accessibility to internal components is required for maintenance, inspection, or replacement. Maximizing welding reduces crevices in the heat exchanger and flanging the connections for accessibility helps reduce maintenance time – both of which support ALARA goals.

#### **9A.2.3.10 Performance and Safety Evaluation**

System reliability is enhanced through the use of periodic system testing, preventive maintenance, and application of redundant instrumentation.

The SDC Safety-Category functions during normal conditions include decay heat removal and, leak detection and isolation as discussed in Subsection 9A.2.3.1.2. There are no Safety-Category functions associated with Off-Normal conditions.

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 (Reference 9A.2.3-1) Chapter 8, Subsection 8.6.6. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

The SDC is located in the Reactor Building which provides adequate protection against natural phenomena ensuring the ability of the SDC to perform its Safety-Category functions.

#### **9A.2.3.11 References**

- 9A.2.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

**Table 9A.2.3-1: Shutdown Cooling System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Process Radiation and Environmental Monitoring System	The Shutdown Cooling System provides a sample line to the PREMS for sampling of reactor coolant when SDC is in service. Radiation Monitor on the SDC overboard line provided by PREMS.	The systems connect downstream of the SDC isolation valve. Radiation Monitor on the SDC overboard line.
Isolation Condenser System	The Isolation Condenser System provides a suction flow path from the RPV via the condensate return lines to the Shutdown Cooling System for decay heat removal and overboarding.	The systems connect downstream of the ICS containment isolation valves.
Control Rod Drive System	The Control Rod Drive system provides water to the Shutdown Cooling System pumps for motor cooling and seal purge flow.	The system interface is on the SDC pump purge water line upstream of SDC isolation valves.
Reactor Water Cleanup System	The Shutdown Cooling System flow can be routed to the Reactor Water Cleanup System for overboarding or into the RPV through the CUW inlet lines for thermal stratification reduction.	The system interface is on the SDC overboard lines at the isolation valve between SDC and CUW.
Liquid Waste Management System	The Shutdown Cooling System can overboard to the Liquid Waste Management System reactor water storage tank if overboard flow is contaminated.	The systems interface on the SDC overboard lines downstream of the isolation valve between SDC and LWM.
Condensate and Feedwater Heating System	The Shutdown Cooling System discharges into the Condensate and Feedwater Heating System lines A and B which discharge inside the RPV to complete a closed cooling loop.	The system interface is on the upstream side of the isolation valve between CFS and SDC.
Main Condenser and Auxiliaries	The Shutdown Cooling System can overboard to the Main Condenser and Auxiliaries if available for overboard flow.	The system interface on the SDC overboard lines downstream of the isolation valve between SDC and MCA.
Plant Cooling Water System	The Plant Cooling Water System provides cooling water to the Shutdown Cooling System heat exchangers.	The system interface is located at the SDC heat exchanger nozzles.
Plant Pneumatics System	The Plant Pneumatics System provides air or nitrogen to operate the Shutdown Cooling System air operated valves.	The system interface is located at each valve actuator.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Safety Class 2 and 3 Electrical Distribution System	The Safety Class 2 and 3 Electrical Distribution System provides power to the Shutdown Cooling System instrumentation and ASD.	The system interface is at each powered component.
Reactor Building Structure	The Reactor Building Structure provides structural support to the Shutdown Cooling System SSCs.	The system interface is at each component support.
Safety Class 1 Instrumentation and Control System	The Shutdown Cooling System includes three divisions of SC1 leak detection instrumentation to provide inputs required by the Safety Class 1 Instrumentation and Control System Distributed Control and Information System.	Input/Output termination cabinets in the Digital Control and Information System cabinet rooms.
Safety Class 2 and 3 Instrumentation and Control System	The Safety Class 2 and 3 Instrumentation and Control System DCIS interfaces with Shutdown Cooling System instrumented and controlled components.	Input/Output termination cabinets in the Digital Control and Information System cabinet rooms.

**Table 9A.2.3-2: BWRX-300 Shutdown Cooling System Mode Table**

Mode	Title	Reactor Mode Switch Position	Shutdown Cooling System Modes
1	POWER OPERATION (10% - 100% Rx Pwr)	RUN	N/A
2	STARTUP	REFUEL <sup>(2)</sup> or STARTUP	Modes A1 and/or B1, B2
3	HOT SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Modes A1 and/or B1, B2
4	STABLE SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Modes A1 and/or B1, B2
5	COLD SHUTDOWN <sup>(1)</sup>	SHUTDOWN	Modes A1 and/or B1, B2
6	REFUELING <sup>(2)</sup>	SHUTDOWN or REFUEL	Modes A1 and/or B1, B2

(1) All reactor pressure vessel head closure bolts fully tensioned

(2) One or more reactor pressure vessel head closure bolts less than fully tensioned



**Table 9A.2.3-3: Shutdown Cooling Configuration During Refueling Mode**

<b>Reactor Water Level</b>	<b>FPC Assist</b>	<b>SDC System</b>	<b>FPC System</b>
Reactor Cavity Pool Filled	Yes – Gate Removed	SDC normally in service (FPC can be evaluated to perform function as needed)	FPC cooling normally in service (SDC can be evaluated to perform function as needed)
Reactor Cavity Pool Filled	No – Gate Installed	SDC train in service (number of trains in service dependent upon the needed cooldown rate/decay heat)	One FPC cooling train normally in service, one in Standby
Reactor Cavity Pool Drained	No – Gate Installed	SDC train in service (number of trains in service dependent upon the needed cooldown rate/decay heat)	One FPC cooling train normally in service, one in Standby

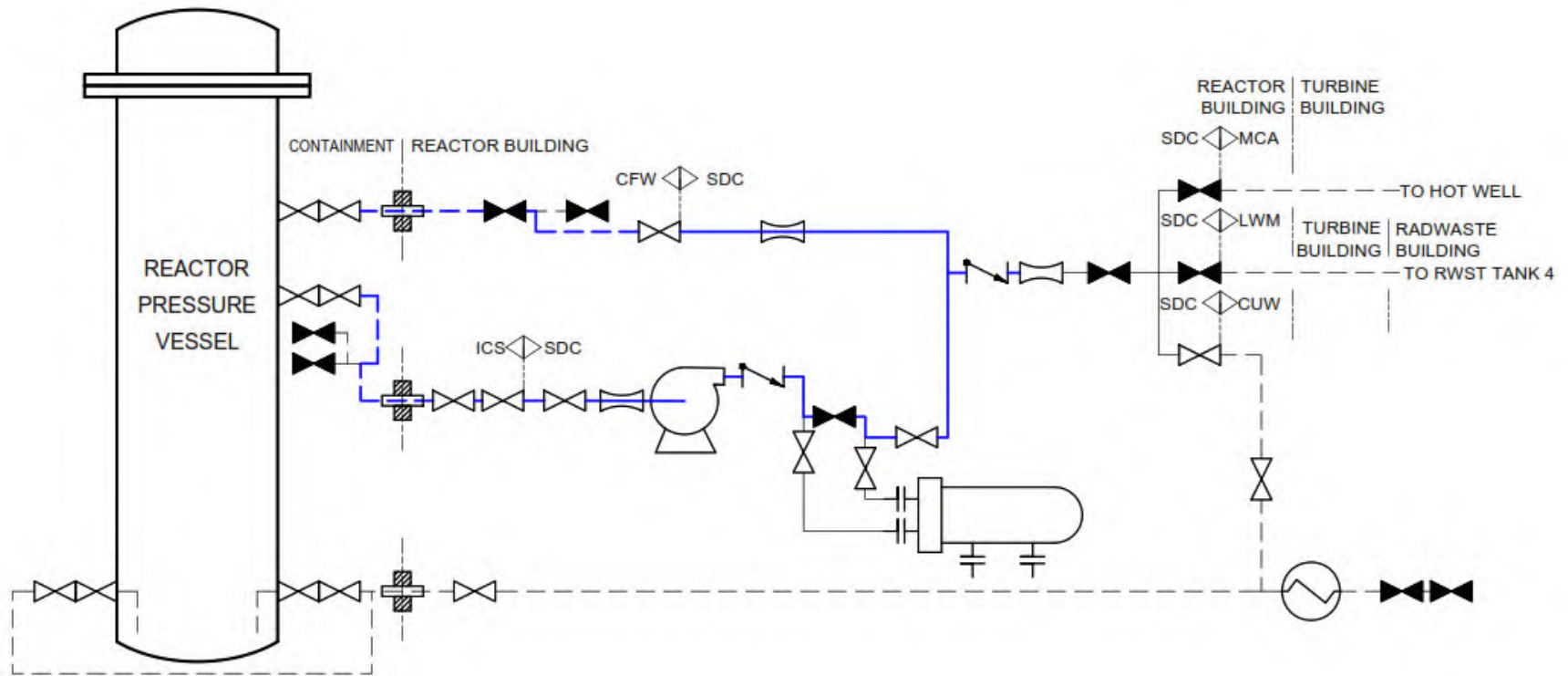


Figure 9A.2.3-1: Decay Heat Removal Subsystem

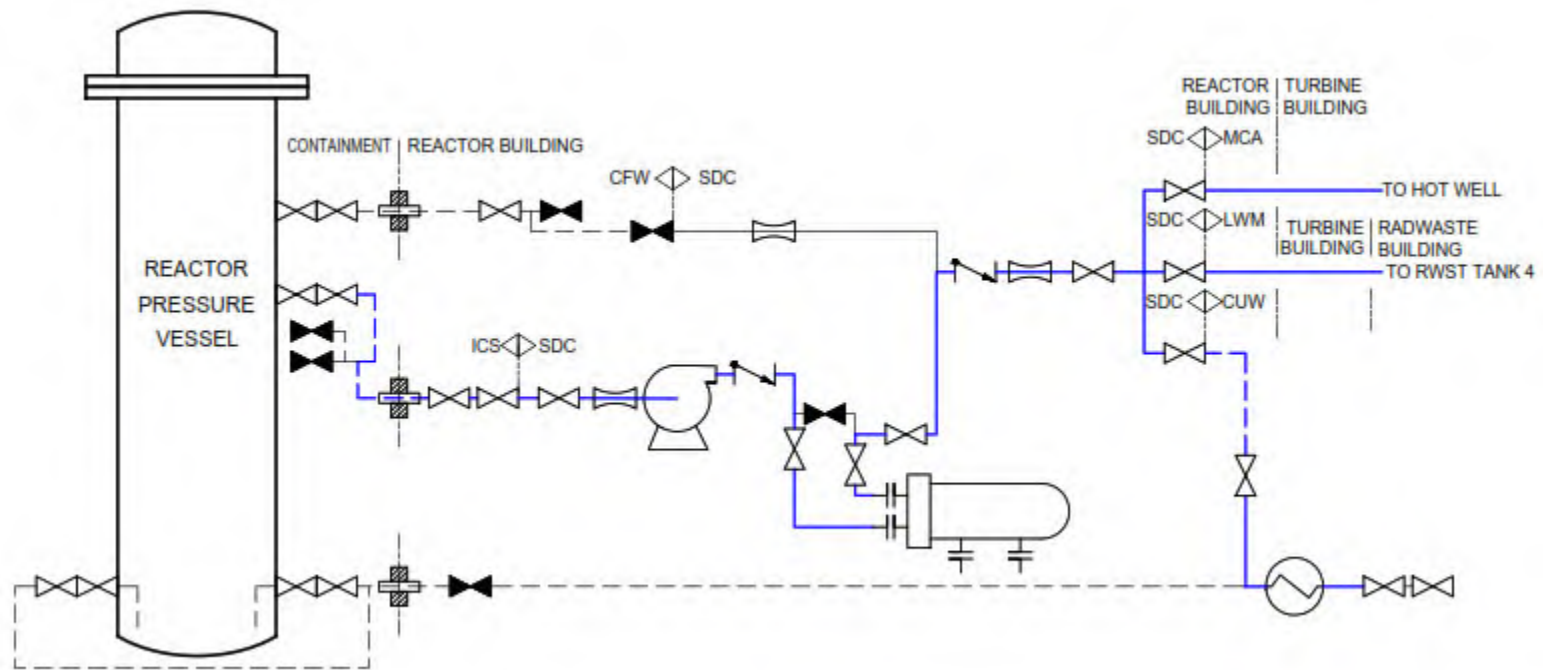


Figure 9A.2.3-2: Overboard Subsystem

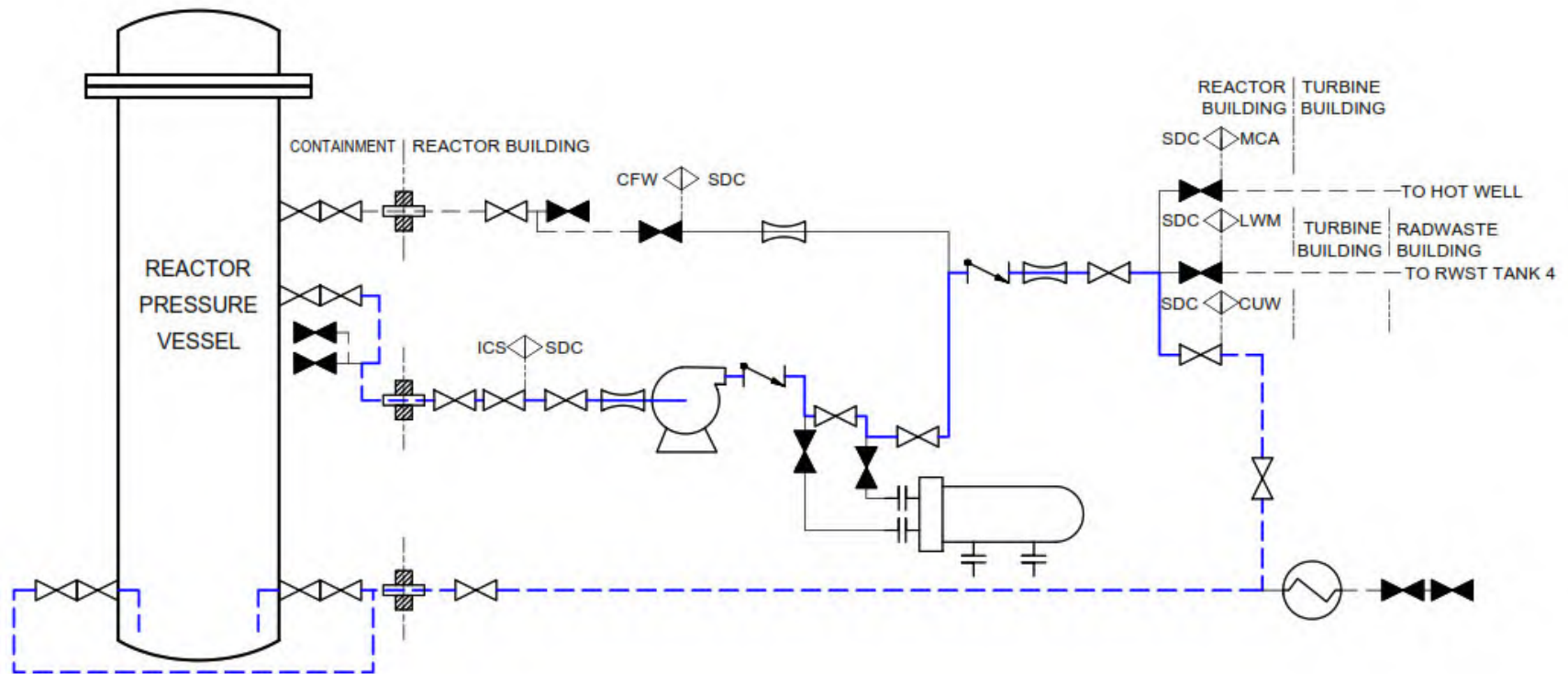


Figure 9A.2.3-3: Thermal Stratification Reduction Subsystem

#### **9A.2.4 Chilled Water Equipment System**

The Chilled Water Equipment (CWE) provides chilled water cooling to the Heating Ventilation and Cooling Systems throughout the plant, to the offgas cooler condenser, to the charcoal adsorber vault Fan Coil Units (FCUs), and to the Containment Cooling System (CCS) in the Reactor Building.

The safety classification of the CWE as well as interfacing SSC is consistent with the requirements of REGDOC 2.5.2, Section 7.1 (Reference 9A.2.4-1).

##### **9A.2.4.1 System and Equipment Functions**

System and equipment functions associated with the CWE are specified as follows. Information pertaining to Defense Lines (DL) is discussed in Chapter 3, Subsection 3.1.6.

###### **9A.2.4.1.1 Normal Functions (Non-Safety-Category)**

CWE provides the following Non-Safety-Category functions:

1. Provides chilled water to cooling coils in HVAC equipment in the following buildings or areas: Control Building (CB), Reactor Building, Turbine Building (TB), and Radwaste Building (RWB)
2. Provides chilled water to selected equipment coolers

###### **9A.2.4.1.2 Normal Functions (Safety-Category)**

The CWE provides the following Safety-Category functions:

1. As part of a Safety-Category (TBD) function, the CWE supplies chilled water to the CCS Air Handling Units (AHUs)
2. The CWE supplies chilled water to the Safety Category (TBD) Instrumentation and Control System, Defense Line 2, Room A, and Safety-Category (TBD) Instrumentation and Control System, Defense Line 2, Room B, Safety Category (TBD) Fan Coil Units
3. The CWE supplies chilled water to the Defense Line 4a Room Fan Coil Units in the Control Building

###### **9A.2.4.1.3 Off-Normal Functions (Non-Safety-Category)**

1. During a LOOP, one chiller/pump set from each chiller train continues to operate and provide cooling to critical Fan Coil Units and AHUs throughout the plant using backup power provided by the standby diesel generators.
2. The rooftop chillers are designed and fabricated to withstand the effects of high winds, earthquakes, and other adverse weather conditions, with the exception of tornado and tornado missiles.
3. During a Toxic Gas Event, the CWE system provides chilled water associated the normal Control Building supply AHU, which continues to operate during the event.
4. In the event of a fire outside the Control Building, the outside air through the normal operating AHU will be secured and the CWE system provides chilled water to operating AHU, which continues to operate in recirculation mode only.

#### **9A.2.4.1.4 Off-Normal Functions (Safety-Category)**

1. As part of Defense Line 3, CWE containment isolation valves are designed to close upon receiving an isolation signal from the Safety Category (TBD) Instrument and Control System.

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

#### **9A.2.4.2 Safety Design Bases**

As part of DL3, CWE containment isolation valves on piping that penetrate the containment boundary are designed to close upon receiving an isolation signal from the Safety Category (TBD) Instrument and Control System.

#### **9A.2.4.3 Description**

CWE is a closed loop chilled water system that supplies chilled water to various AHU cooling coils and plant equipment coolers in the Turbine Building, Radwaste Building, Reactor Building, and Control Building. Heat absorbed by the CWE is rejected from the CWE chiller condensers mounted on the Radwaste Building roof to atmosphere.

CWE is comprised of four (4) CWE air-cooled chillers, four (4) CWE pumps, two (2) expansion tanks (one per train), four (4) air separators, one (1) chemical bypass feeder, one (1) glycol auto-fill unit, piping, valves, instruments, and controls.

The CWE is split into two trains which allows for the isolation of the two different sets of AHUs and FCUs supporting redundant defence-in-depth equipment. Each CWE train consists of two chillers, two pumps, two air separators, and an expansion tank, while the glycol auto-fill unit and chemical bypass feeder are shared among both trains.

Both CWE trains are cross-tied together with air operated valves, which are normally open, to allow for the three active chillers to evenly share the heat load; however, these valves can be closed for train separation.

Figure 9A.2.4-1 depicts a simplified flow diagram of the CWE System.

Four 33 1/3% capacity air-cooled chillers are provided to reject heat from the closed chilled water loop to the environment. During normal operation, three chillers are in service while the fourth is in standby mode. Each chiller consists of an evaporator section, condenser section, refrigerant compressor, controls, and an integrated waterside economizer which allows for the chillers to run at reduced electrical loads at lower ambient temperatures. The CWE chillers utilize R-134a to exchange heat with the propylene glycol/water mixture. Since these units are mounted outside, the use of R-134a does not pose a risk to plant personnel. Each CWE chiller is provided with built-in protection against freezing.

The four 33 1/3% capacity in-line chiller pumps deliver chilled water from the CWE return header to the dedicated chiller. During normal operation, three pumps are in service while the fourth is in standby mode. Each pump discharge line is provided with a check valve to prevent backflow through the pump.

The CWE system trains are powered by separate generator load groups as part of the defense in depth protection function.

Four air separators are provided to remove entrained air. An air separator is connected to the chilled water return header upstream of each chiller pump. The air that is removed by the air

separators is piped to the train specific expansion tank to provide the gas cushion for chilled water expansion and contraction.

The two expansion tanks are connected to the two main chilled water return lines and provide a reservoir of demineralized makeup water to account for small amounts of system leakage and accommodate thermal expansion and contraction of water within the system while maintaining the system pressure. The expansion tank level is also used to determine the need for additional makeup fluid. Each expansion tank is shrouded to prevent rain, snow, and other debris from collecting in the expansion tank curbed area. The shroud is fitted with sight glasses for operators to manually inspect the curbed area for any liquid that has been collected.

The glycol auto-fill unit has a connection to the Water, Gas, Chemical Pads System (Subsection 9A.9.5) to provide demineralized water to maintain a predetermined propylene glycol/water mixture.

Isolation valves are provided at the interfaces with the components being cooled by CWE to allow for maintenance on those items without impact to the CWE system operation.

The supply and return line penetrations have isolation valves with electrical division 1 and division II connections outside and inside containment respectively, and ASME Class 2 piping into containment. The return piping inside of primary containment is provided with relief valves to protect against overpressure caused by thermal expansion. The system is designed to meet mechanical and electrical separation and redundancy requirements to ensure that the consequences of a single failure, pipe whip, jet forces, missiles, and the effect of failure of any non-Seismic Category I components do not compromise the ability of the system to meet its safety category requirements.

The CWE containment isolation valves are designed to operate in environments associated with the normal and accident conditions in the RB to which they are exposed. Refer to Chapter 3, Subsection 3.9.4 for information pertaining to Environmental Qualification.

The CWE expansion tanks, glycol auto-fill unit, chemical bypass feeder, and chillers pumps are all surrounded by a permanent curb to prevent the accidental excursion of propylene glycol into the Equipment and Floor Drainage System (Subsection 9A.9.3) sumps. To minimize dose exposure to plant personnel, the isolation valves for the FCUs located inside the bioshield area of the Turbine Building are positioned outside the bioshield area.

#### **9A.2.4.3.1 Component Description**

##### **Air-Cooled Chillers**

Four (4) 33% capacity air-cooled chillers are provided. Each chiller consists of an evaporator section, condenser section, refrigerant compressor, and controls. The chillers have individual temperature controls.

The CWE chillers are tested in accordance with Air Conditioning and Refrigeration Institute (AHRI) Standard 550 (Reference 9A.2.4-2) and American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) Standard 30 (9A.2.4-3).

Refrigerant piping is designed and fabricated per CSA B52 (Reference 9A.2.4-5).

Each chiller is equipped with pressure relief valves which relieves excess pressure of the refrigerant charge to the atmosphere in accordance with ASHRAE 15 Safety Code and ASME BPVC, Section VIII.

The chiller capacities include a 15% margin to accommodate unaccounted hot surfaces, higher than anticipated latent loads, and other unknown cooling loads

## **Pumps**

Four (4)x33% capacity centrifugal, electric motor driven pumps are provided with variable frequency drivers.

Pumps are factory tested per ANSI/HI 14.6.

The pumps are sized such that two (2) pumps are capable of supplying the necessary flow during a Loss-of-Offsite Power.

## **Chemical Bypass Feeder**

A carbon steel chemical bypass feeder tank is provided to treat the glycol water mixture and prevent the development of organics within the closed cooling water loop.

The bypass feeder tank is designed to meet the requirements of ASME BPVC, Section VIII.

## **Expansion Tanks**

The two atmospheric vessels are designed to meet the requirements of API 650.

## **Glycol Auto Fill Unit**

The packaged, glycol auto fill unit maintains the glycol system pressure by providing glycol make-up automatically upon a drop in system pressure. The glycol auto-fill unit has a connection to the Water, Gas and Chemical Pads system to provide demineralized water to maintain a predetermined propylene glycol/water mixture. Upon a drop in liquid level in the expansion tanks, the integrated side suction peripheral pump starts adding fluid from the glycol auto fill tank to the CWE System expansion tanks until the setpoint level is reached.

## **Piping and Valves**

All non-refrigerant piping is designed in accordance with the requirements of ASME B31.1 with the exception of the containment penetration portion. The containment isolation valves and lines that supply and return chilled water to the containment cooling system cooling coils are Safety Class 1, designed to the requirements of BPVC Section III, Division 1- Subsection NCD; and are Seismic Category I. The containment isolation valves are designed to fail-closed on a loss of air or signal.

Refrigerant piping inside the air-cooled chillers is designed and fabricated in accordance with the requirements of ASHRAE 15(Reference 9A.2.4-2).

Overpressure relief valves are provided on the CWE return lines located both inside and outside of containment.

CWE piping is provided with insulation to prevent pipe sweating as required. The pipe insulation meets the combustibility requirements in compliance with fire protection codes and standards.

### **9A.2.4.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

Materials are selected in accordance with applicable codes, standards, and industry practice for the design, service, and test conditions and expected ambient conditions. Materials are compatible with the internal process and external environmental conditions during normal, abnormal, accident, and beyond design basis accident conditions as appropriate. Building



construction utilizes noncombustible materials as defined in the fire protection Codes and Standards.

#### **9A.2.4.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.4-1 for Chilled Water Equipment interfaces with other equipment or systems.

#### **9A.2.4.6 System and Equipment Operation**

##### ***9A.2.4.6.1 Normal Operational Concept***

Up to three of the four chiller pump sets are placed in normal operation based on the cooling load demand. One chiller pump set is always available in standby. The CWE is designed to operate with N+1 redundancy. Where N is the number of pieces of equipment to meet the baseload demand. N+1 indicates there is one extra piece of equipment provided. Hence during normal operation 3 chillers/pumps will be in service while the 4th be in standby mode.

The chilled water control valves modulate in response to room or supply air temperature controllers for the Control Building and Reactor Building, which are part of the HVS except for the Containment Cooling System (Subsection 9A.5.6) AHUs. The temperature control valve downstream of the CCS AHU cooling coils modulate in response to the air discharge temperature of the CCS AHUs. As the cooling loads decrease, the valves modulate to decrease the chilled water flow causing a differential pressure increase across the supply and return mains.

Chillers are automatically started or stopped, based on the CWE supply header temperature.

The glycol auto-fill unit will automatically be switched on and off based on the level transmitter in the expansion tanks. Upon receiving a low level signal, the glycol auto-fill unit will switch on and add additional fluid to the CWE expansion tanks.

Normally, chiller train A, which consists of chillers and pumps 1 and 2, and expansion tank A; and train B, which consists of chillers and pumps 3 and 4 and expansion tank B are cross-tied together such that flow from the RWB, TB, TB switchgear rooms A and B, RB loops A and B, and CB loops A and B, will be shared proportionally between the two trains based on the number of active chillers in each train.

Depending on the ambient conditions, the chillers are able to operate in one of three different modes: mechanical cooling only, hybrid cooling, or free cooling only. When the ambient temperature is too warm to provide free cooling, an integrated three-way valve at the chiller inlet allows the glycol/water mixture to only run through the condenser coils. When the temperature reaches an ambient temperature where some free cooling is feasible, the three-way valve diverts some flow through the free cooling coils which then enters the chiller evaporator. If further fluid temperature reduction is required, the chillers will perform the remaining mechanical cooling. As the ambient temperature continues to decrease, there will be a point where the supply temperature can be met by free cooling operation alone and the mechanical cooling can be shut off completely. During free cooling mode, the chillers operate using significantly less power because the waterside economizer does not use a refrigerant loop and the section requires fewer hot air discharge fans. When one or multiple FCUs located inside the Turbine Building bioshield area are manually isolated because of a leak, the chilled water trapped between the isolation valves is drained to prevent damage to the CWE distribution pipe or FCUs tubes due to thermal expansion.

##### ***9A.2.4.6.2 Off-Normal Operational Concept***

During a LOOP and turbine trip, one CWE chiller and pump from each chiller train will trip and restart upon receiving power from the standby diesel generators. The inlet guide vane of the non-

operational chillers, and the block valves in the TB, excluding the TB switchgear rooms, and RWB return headers close automatically, so that the CWE flow will only flow through the equipment required for the continued safe operation of the plant.

Upon station blackout, the CWE is not operating.

#### **9A.2.4.7 Instrumentation and Control**

##### **9A.2.4.7.1 Instrumentation**

CWE contains sufficient instrumentation to:

1. Verify chilled water pump performance, by monitoring pump suction and discharge pressure.
2. Verify chiller performance, by monitoring the inlet and outlet pressure, CWE flowrate, and the inlet and outlet temperatures for each chiller.
3. Verify chilled water return and supply header temperatures.
4. Verify pressure loss through the system.
5. Provide remote status indication of the air-operated containment isolation valve to the MCR and SCR.

##### **9A.2.4.7.2 Controls**

System and component operating status, including the state of manual overrides, and the state of the A and B trains are provided at the Main Control Room (MCR) and Secondary Control Room (SCR). Manual initiation and shutdown of CWE is provided from the MCR.

The following CWE primary displays and alarms are provided:

1. MCR and SCR Major Indications:
  - a. Chiller operating status
  - b. Chiller inlet and outlet pressures and temperatures
  - c. CWE system operating pressures and temperatures
  - d. CWE pump suction and discharge pressures
  - e. CWE pump flow rates
  - f. Air-operated containment isolation valve status
  - g. Expansion tank liquid level
  - h. CB, RB, and TB switchgear room loops misaligned (only possible when the chiller headers are isolated)
2. MCR and SCR Alarms:
  - a. Loss of CWE flow to any operating chiller
  - b. Chiller trip and chiller trouble
  - c. High CWE return water temperature
  - d. High or low chiller outlet temperature
  - e. High or low expansion tank liquid level

The CWE is expected to run continuously with three of the four chillers and pumps running while the other chiller and pump are on standby. The operator manually brings the chiller/pump combo online by selecting an operating chiller to put into standby. Upon receiving the standby command, the standby pump will start, followed by the chiller. Once the MCR or SCR receives confirmation that the pump and chiller are both on, the operating chiller will stop, followed by the pump. The chillers and chiller support systems are capable of automatically starting and stopping based on the CWE supply header temperature; however, the chillers and CWE pumps can be operated remotely from the MCR.

A chilled water temperature control valve located in the return piping at each AHU cooling coil bank, is designed to maintain air discharge temperature of the upstream AHU. The chilled water control valves modulate in response to room or supply air temperature controllers, which are part of the HVS except for the CCS AHUs. The temperature control valves downstream of the CCS AHU cooling coils modulate in response to the air discharge temperature of the CCS AHUs. As the cooling loads decrease, the valves modulate to decrease the chilled water flow causing a differential pressure increase across the supply and return mains.

The automatic block valve in each building's chilled water return line is modulated, during plant startup, to balance pressure loss in each building chilled water loop so that flow is adequately distributed to every building.

The temperature control valves in each building's chilled water return line can be modulated to balance pressure loss in each building's chilled water loop so that flow is adequately distributed to every building.

During normal operation, the chillers operate to automatically maintain the chilled water supply temperature at a predetermined value.

The standby chiller pump will auto start upon trip of any operating pump. The CWE pump is tripped on occurrence of any of the following:

- Low CWE pump suction pressure
- A manual stop command
- A protective device trip

The chiller is tripped on occurrence of any of the following:

- Loss of CWE flow
- A failure of the chiller automatic start sequence
- A manual stop command
- High system pressure

The containment isolation valves are automatically closed upon receiving a containment isolation signal and also automatically fail-closed on loss of air or signal.

#### **9A.2.4.8 Monitoring, Inspection, Testing, and Maintenance**

Pre-operational testing is performed in accordance with applicable codes and manufacturer recommendations.

The CWE chillers are tested in accordance with AHRI Standard 550 (Reference 9A.2.4-3) and ASHRAE Standard 30 (Reference 9A.2.4-4).

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package and are a part of a maintenance program conforming to ASME Code OM 2020.

Areas requiring inspection are provided with access and removable insulation.

CWE piping and valves for the containment penetrations are tested in accordance with 10CFR50, Appendix J and CNSC REGDOC-2.5.2. Test and vent connections are provided at the containment isolation valves in order to verify that the valves meet the local leak rate limits.

The containment isolation valve closure time is monitored during the valve operability test and the leakage is monitored or verified during the valve leakage test as specified in the containment leakage testing program. Leak detection and inspection for primary containment isolation features is designed to ASME BPVC Section III, Division 1, Class 1. A test connection is provided to support local leak rate testing of the primary containment boundary. The test connections minimize the amount of water that must be drained to permit periodic testing of containment isolation valves that include operability testing, leak rate testing, valve status verification, and test frequency.

Containment isolation valve testing, stroke time testing, and leakage rate testing are incorporated to ensure proper and safe functionality of the valves.

Isolation valves are provided at each piece of equipment, control valve, and piping circuit so they can be isolated for maintenance and repair.

Chillers and pumps are arranged to provide adequate floor space and unobstructed clearance to permit monitoring, maintenance, and inspection of each unit.

Isolation valves are provided at the interfaces with the components being cooled by CWE to allow for maintenance on those items without impact to the CWE system operation.

#### **9A.2.4.9 Radiological Aspects**

Chapter 12 provides information pertaining to design measures that are taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA).

#### **9A.2.4.10 Performance and Safety Evaluation**

The CWE is split into two trains which allows for the isolation of the two different sets of AHUs and FCUs supporting redundant defense in depth equipment.

CWE is designed with sufficient redundancy to assure that chilled water is normally available during all modes of plant operation, including startup and shutdown. Certain equipment needed during a Loss-of-offsite power are provided Diesel Generator backup power.

As part of DL3, the CWE performs a containment isolation Safety-Category function. The CIVs are designed to close upon receipt of an isolation signal from the Safety Category (TBD) Instrument and Control System. During normal operations the CWE supplies chilled water to the CCS AHUs, DL2 Room A and Room B FCU's, and chilled water to the DL 4a Room FCU in the Control Building.

#### **9A.2.4.11 References**

- 9A.2.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.4-2 ASHRAE Standard 15, "Safety Standard for Refrigeration Systems."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.2.4-3 AHRI Standard 550/590 (I-P), "Performance Rating of Water-chilling and Heat Pump Water-heating packages using the Vapor Compression Cycle," Air Conditioning, Heating, & Refrigeration Institute.
- 9A.2.4-4 ASHRAE Standard 30, "Method of Testing Liquid Chillers."
- 9A.2.4-5 CSA B52, "Mechanical Refrigeration Code," CSA Group.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.2.4-1: Chilled Water Equipment System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Safety Category (TBD) Instrument and Control System	Provides all Safety Category (TBD) I&C control for containment isolation valve logic	At Input/Output termination
Safety Category (TBD) Instrumentation and Control System	Provides Safety Category (TBD) I&C controls including pump logics, valve logics, excluding the CIV, instrumentation (pressure, flow, temperature, level) etc.	At Input/Output termination
Non-Safety Instrumentation and Control System	Provides Non-Safety-Category controls	At Input/Output termination
Offgas System	CWE provides chilled water for the offgas cooler condenser and charcoal adsorber vault FCU cooling coils	Offgas cooler condenser flanges and isolation valves for the charcoal adsorber vault FCU
Plant Pneumatics System	PPS provides instrument air/nitrogen to pneumatically operated valves	At CWE air operated valves
Safety Category (TBD)Electrical Distribution System	Provides normal and standby diesel backed power to the CWE chillers and pumps	At CWE equipment
Non-Safety Electrical Distribution System	Provides Non-Safety-Category power to CWE equipment	At CWE equipment
Containment Cooling System	CWE provides chilled water for containment cooling system chilled water coil loads	CCS AHU flanges
Heating, Ventilation, and Cooling System	CWE provides chilled water for the HVS chilled water cooling coil loads	Isolation valves at AHU and FCU
Equipment and Floor Drain System	The condensate collected from the pump baseplate drain is piped to the EFS sumps. The liquid connection from the pressure relief valves inside containment discharge to the containment sump, while the pressure relief valve in the RB, outside of containment, is piped to the floor drain	Drain pipe from chiller pumps and liquid connection from RB PRVs
Water, Gas, Chemicals Pads	Water, Gas, Chemicals System supplies makeup water to the glycol auto-fill unit	Isolation valve at glycol auto-fill unit

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Process Radiation and Environmental Monitoring System	PREMS provides process sampling for chemistry monitoring	Grab sample station(s)

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

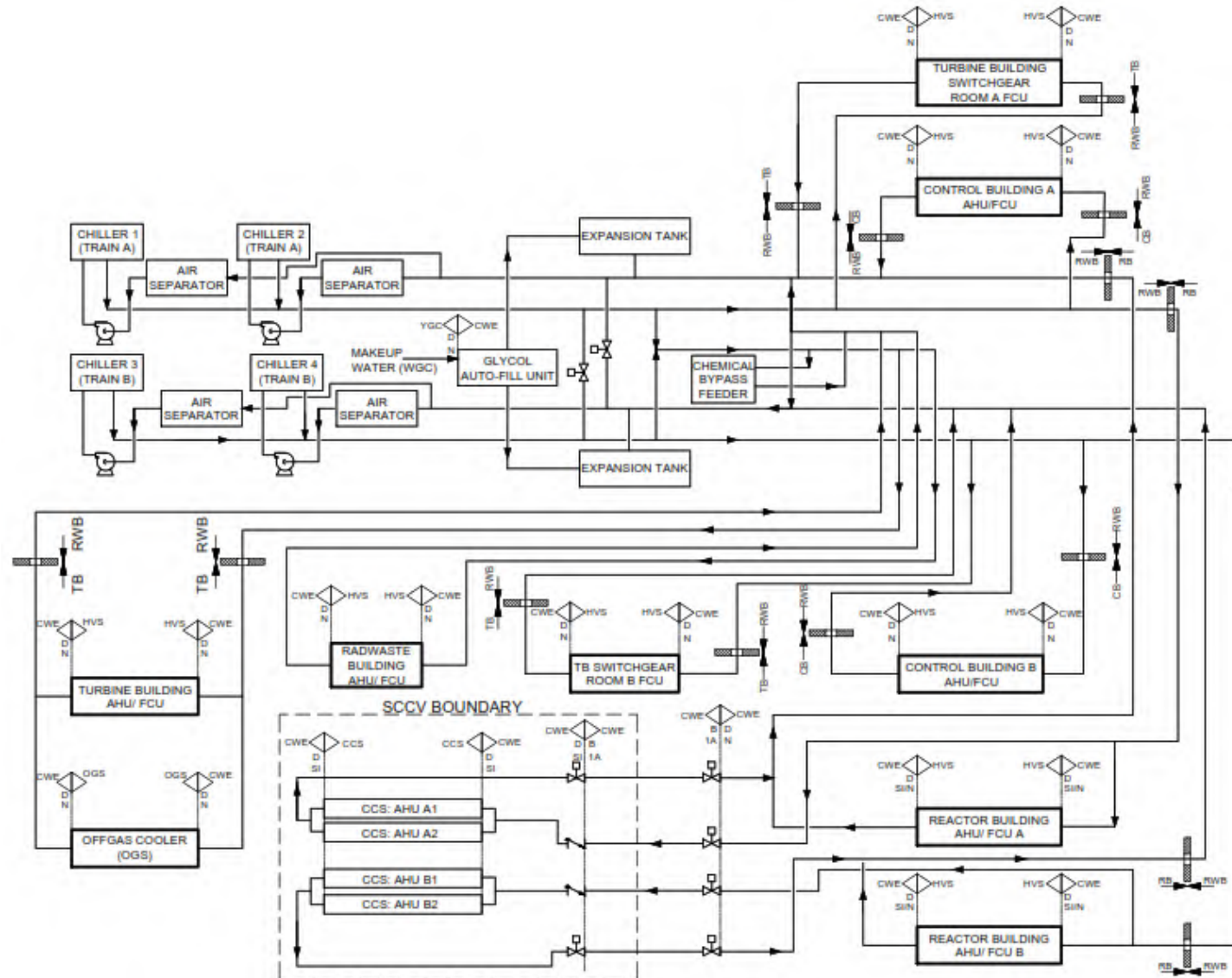


Figure 9A.2.4-1: Chilled Water Equipment



### **9A.2.5 Normal Heat Sink System**

The Normal Heat Sink (NHS) provides heat rejection for the CWS and the PCW. Refer to Chapter 10, Sections 10.8, and Subsection 9A.2.1 for information related to the CWS and PCW respectively.

Refer to Subsection 9B.3.5 for information pertaining to the Pumphouse, Forebay, and Tunnels.

#### **9A.2.5.1 System and Equipment Functions**

##### **Normal Functions (Non-Safety-Category)**

The NHS provides the supply of cooling water for the circulating water pumps, which removes heat from the condenser and PCW heat exchangers and accepts the circulating water return flow. NHS performs this function at all times for all plant modes of operation.

##### **Normal Functions (Safety-Category)**

The NHS performs the following Safety-Category functions during normal conditions:

- Provides cooling water as a means to reject heat from the MCA to the environment
- Provides cooling water as a means to reject heat from the PCW heat exchangers to the environment

##### **Off-Normal Functions (Non-Safety Category)**

The system does not perform any Non-Safety-Category functions during off-normal conditions.

##### **Off-Normal Functions (Safety-Category)**

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

#### **9A.2.5.2 Safety Design Bases**

The NHS provides cooling water to the two CWS (Chapter 10, Subsection 10.8) plant cooling water supply pumps (2A and 2B), which in turn are relied upon to provide cooling water to the PCW heat exchangers, to support Safety-Category functions for FPC (Subsection 9A.1.3) and SDC (Subsection 9A.2.3).

#### **9A.2.5.3 Description**

The NHS is a once-through cooling water system design. Water will flow into the intake structure forebay from Lake Ontario by means of an intake tunnel. The water is strained prior to pumping by the CWS pumps. Water flows through the CWS system where it absorbs the heat from the main condenser and the PCW heat exchangers. The water is discharged back into Lake Ontario through the discharge structure. A recirculation line from the CWS discharge line to the intake is provided to moderate NHS temperature at the Forebay of the NHS, as needed, during cold weather conditions.

#### **9A.2.5.4 Materials**

Material and process control requirements for the BWRX-300 structures and components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from corrosion (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

#### **9A.2.5.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.5-1 for system interfaces.

#### **9A.2.5.6 System and Equipment Operation**

The NHS system operates under all normal operating modes of the plant. The Off-normal operational modes include operations following a LOOP event.

##### ***9A.2.5.6.1 Normal Operational Concept***

The NHS has its water pumped by the circulating water pumps to the circulating water supply in the Turbine Building where it absorbs the heat from the main condenser and the PCW heat exchangers. Next, the CWS dedicated PCW and Condenser return lines discharge the circulating water back to the heat sink.

##### ***9A.2.5.6.2 Off-Normal Operational Concept***

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

#### **9A.2.5.7 Instrumentation and Control**

The NHS has no instrumentation and controls associated with the system. All NHS parameters are measured within CWS.

#### **9A.2.5.8 Monitoring, Inspection, Testing, and Maintenance**

Maintenance and inspection is based upon integrating proactive, reactive, preventive, and predictive maintenance and operating experience. Implementation of maintenance and inspection increases the probability that the NHS structures, systems, and components function in the required manner over their design life cycle. Operational testing is performed in accordance with plant procedures.

#### **9A.2.5.9 Radiological Aspects**

Chapter 12, Subsection 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.2.5.10 Performance and Safety Evaluation**

The NHS is capable of performing its design functions during all modes of operation.

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

**Table 9A.2.5-1: Normal Heat Sink Interfaces**

Interfacing System	Interface Description	Interface Boundary
Circulating Water System	The NHS provides the source of water used as the cooling medium to the CWS pumps and accepts the return flow from the CWS pumps. When required for cold weather operation, CWS is designed to maintain a minimum intake basin temperature through a recirculation line from the CWS discharge line	At the intake and discharge structures

### **9A.2.6 Isolation Condenser System Pool Cooling and Cleanup System**

The primary function of the Isolation Condenser System Pool Cooling and Cleanup System (ICC) is to remove heat from the Isolation Condenser System (ICS) (Chapter 6, Subsection 6.2) pools such that the bulk temperature of water in the pools is maintained below Technical Specifications limits and thereby ensure the readiness of the ICS to perform its Safety-Category function. Secondary functions of the ICC include maintaining the cleanliness of the ICS pool water and providing the capability to add clean makeup water during normal reactor operations to offset the minor routine loss of water inventory due to evaporation.

#### **9A.2.6.1 System and Equipment Functions**

The ICC performs the following functions during normal and off-normal conditions.

##### **9A.2.6.1.1 Normal Functions (Non-Safety-Category)**

1. The ICC extracts heat from the ICS Pool Water to maintain IC (Isolation Condenser) pool temperature within TS limits for maximum pool temperature, and thereby maintains ICS pool readiness for IC deployment.
2. The ICC removes soluble and insoluble impurities from the ICS Pool Water to comply with plant water quality requirements.
3. The ICC provides the capability of adding makeup demineralized water to the ICS pools during normal reactor operations to replace water inventory lost by evaporation from the ICS pool surface.

##### **9A.2.6.1.2 Normal Functions (Safety-Category)**

The Safety-Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools. The Suction Surge Tank is one of two primary interfaces with the Reactor Building Structure (RBS) ICS Pool Structure and is categorized as SC1 and as a Seismic A component. The Return Guard Pipe is the second of two primary interfaces with the RBS ICS Pool Structure and is likewise categorized as an SC1 and Seismic A component. The principal function of these two components is to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools.

##### **9A.2.6.1.3 Off-Normal Functions (Non-Safety-Category)**

The ICC does not provide any Non-Safety-Category functions during off-normal conditions.

##### **9A.2.6.1.4 Off-Normal Functions (Safety-Category)**

The ICC does not provide any Safety-Category functions during off-normal conditions.

The design of the ICC meets requirements specified in CNSC REGDOC-2.5.2 Section 7.1 (Reference 9A.2.6-1) as related to ensuring that the portions of the ICC, namely the suction and return pipe penetrations into the ICS pools, that interact with the SC1 ICS are also designed to SC1 requirements. Considering that the ICC is of a lower safety classification than the ICS, designing the suction and return pipe penetrations to SC1 requirements ensures that the failure of a lower safety class SSC cannot propagate to an SSC belonging to a higher safety class.

#### **9A.2.6.2 Safety Design Description**

The Safety-Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools.

The Suction Surge Tank and the Return Guard Pipe are designed to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools. The probability of such

breaks occurring is minimized by locating ICS pool isolation valves as close to the ICS pool penetrations as practicable. In the event of a non-isolable break in the ICC suction piping, it is possible for the ICS Cubicles (i.e., Inner Pools) to drain to the bottom of the connecting weir in the partition wall separating the ICS Cubicles from the ICS Outer Pools, and for the ICS Outer Pools to drain to the top of the Suction Surge Tank. In the event of a non-isolable break in the ICC return piping, the design of the Return Guard Pipe prevents draining the ICS Outer Pools and Cubicles because the top of the Return Guard Pipe is above the maximum pool water level.

ICC suction and return connection interfaces with the Reactor Building Structure are designed such that any failure in the ICC pressure boundary cannot result in draining the ICS Pools.

#### **9A.2.6.3 Description**

The ICC is classified as Non-Safety Class, Non-Seismic Category and Quality Group D. The ICC has no Defense Line Functions. Figure 9A.2.6-1 depicts the ICC.

The ICC consists of two independent and identical 50% capacity trains which service the three Isolation Condenser Cubicle Pools. The two identical thermal processing trains can be operated together or operated with one train and the other train shutdown as needed to maintain conditions in the ICS pools.

Both ICC trains take suction from a single penetration in the Isolation Condenser Outer Pool A. An air-operated ICS Pool Isolation Valve is installed in the main suction pipe to stop flow from Outer Pool A. Processed ICS pool water is returned directly to the three IC cubicles. Return water can be directed to individual IC cubicles by controlling the flow of ICC effluent by aligning IC cubicle isolation valves as desired. The ICS Pool suction and return isolation valves are equipped with manual overrides and handwheels, which allows the valves to be manually operated if the valve actuator fails, thereby increasing system reliability.

Each ICC pump/heat exchanger train can be isolated as needed to maintain ICS readiness to allow both trains to operate simultaneously, or so that one train can operate separately to allow maintenance to be performed on the other train for continuous operation. Each ICC train includes a centrifugal pump with Adjustable Speed Drive (ASD), frame-and-plate HX, associated piping, sensors, and valves. PCW (Subsection 9A.2.1) flows through the HX to extract heat from the ICS pool water which is rejected to the environment. The discharge from the two ICC pump/HX trains is processed by a skid-mounted demineralizer before being returned to the IC Cubicle Pools. All ICC flow passes through the demineralizer except when bypassed. The demineralizer can be bypassed when the temperature of the discharge exceeds the maximum temperature allowed for the demineralizer resins or when chemicals are injected into the water to inhibit corrosion and biological growth in the IC pools and ICC components. The ICC is equipped with a dosing pot for injecting chemicals for corrosion and biological control into the process fluid.

Because the ICC is of a lower safety class and quality group than the ICS and pools, design provisions are provided to ensure that a pressure boundary failure in its piping or components cannot adversely impact the higher safety class system, specifically draining of the ICS pools. This is accomplished by a surge volume (Suction Surge Tank) that is a component of the seismically designed reactor building IC pools structure, from which the ICC takes suction. In the event of a failure in the ICC that creates a drainage path, only the surge volume can be drained with no impact on the safety class IC pools volume, and with minimal impact on the ICC components from flooding. Similarly, the discharge or return piping to the IC pools is routed through the seismic Return Guard Pipe, and then routed near the ceiling of the ICS Outer Pools and across the ICS cubicle wall through the connecting weir at which point the return pipes are submerged and routed to a distribution sparger that discharges the cooled and purified water

directly to the ICS Cubicle near the bottom of the IC. The high point of the return piping contains an anti-siphon feature to prevent backflow in the event of an ICC pressure boundary failure. Additionally, ICS pool isolation valves are located as close as possible to the pipe penetrations to minimize the impact to the ICS pools for any failure of ICC piping beyond the isolation valves.

#### **9A.2.6.3.1 Component Description**

The following information is provided relative to the major components in the ICC system.

##### **Suction Surge Tank**

The Suction Surge Tank is one of the primary interfaces with the ICS Pools and, therefore, is an ICC component with SC1 and Seismic A classifications. The Suction Surge Tank consists of flanged (removable) pipe spools. All Suction Surge Tank components are made of Type 304/304L or 316/316L stainless steel. The main suction pipe penetration into the ICS Outer Pool A inside the Suction Surge Tank is designed to provide a leakage-tight barrier between the ICS Outer Pool A and the underlying ICC Equipment Area. A cover plate with vent tube for the Suction Surge Tank is provided to permit the Suction Surge Tank to be drained for downstream maintenance activities without draining the ICS Outer Pools.

##### **Return Guard Pipe**

The Return Guard Pipe is the second of two primary interfaces with the ICS Pools and, therefore, is an ICC component with SC1 and Seismic A classifications. The Return Guard Pipe consists of flanged (removable) pipe spools. Return Guard Pipe components are made of Type 304/304L or 316/316L stainless steel. The return pipe penetration into the ICS Outer Pool A, inside the Return Guard Pipe is designed to provide a leakage-tight barrier between the ICS Outer Pool A, and the ICC Equipment Area. The Return Guard Pipe includes an attached cover that seals the Return Guard Pipe to prevent water accumulation and debris from entering the Return Guard Pipe.

The Return Guard Pipe cover plate assembly is designed to provide lateral support for the anti-siphon devices in the return piping to meet seismic structural requirements.

##### **Filtration**

A cylindrical metallic screen is installed at the main suction inlet for foreign material exclusion. The foreign material exclusion screen is connected to the main suction pipe by a flanged connection. The foreign material exclusion screen is designed to facilitate periodic cleaning to remove accumulated debris on the outer surface of the screen. In-line filters with replaceable filter cartridges are installed at the inlets to the heat exchangers on both the hot side and the cold side to remove minute suspended particulates to prevent clogging the narrow flow channels between plates in the heat exchangers for maximum heat transfer effectiveness. All filtration components and surfaces in contact with the process fluid is made of Type 304/304L or 316/316L stainless steel.

##### **Adjustable Speed Drive Pump Motors**

The ICC pumps are driven by ASDs with electric motors to provide operational flexibility for: a) optimal performance in restoring ICS Pool Water temperature to TS requirements following deployment of one or more ICs; and b) efficient and economical performance in maintaining ICS Pool Water temperature and cleanliness during normal reactor operations.

##### **Pumps**

Pump suction and discharge connections are flanged. All pump components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. The ICC pumps are designed to comply with the requirements of ANSI/HI 1.3 (Reference 9A.2.6-2). Each

pump is powered by an Adjustable Speed Drive with electric motor. The pumps have mechanical seals to minimize the potential for leakage and reduce the need to perform maintenance on shaft seal packings. Pumps are installed within catch basins to contain minor leakage. Moisture sensors are installed in the catch basin to inform reactor operators of potential maintenance issue due to water leakage.

### **Heat Exchangers**

Frame-and-plate HXs are used to reject heat from the process fluid in the ICC loop to the PCW. HX components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. HX inlet and discharge connections are flanged to facilitate removal as necessary for maintenance, repair, and replacement. Drainage taps are provided in HX or piping connections to allow the HX to be drained to facilitate maintenance or repair activities.

### **Demineralizer**

An integrated skid-mounted demineralizer is used to clean the process fluid in the ICC loop. Demineralizer components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. A drainage tap is provided to allow the Demineralizer tank to be drained to facilitate maintenance or repair activities. The Demineralizer is sized to process the effluent from a single ICC pump train. ICC Demineralizer spent resin media is discharged to the Solid Waste Management System (Chapter 11, Section 11.4).

### **Dosing Pot**

A dosing pot allows chemical injection into the process fluid to inhibit corrosion and biological growth in ICC piping and components and the ICS pools. All dosing pot components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. A drainage tap is provided to allow the dosing pot to be drained when not in use.

### **Backflow Preventer**

Back flow preventers are installed in the return piping to the IC cubicles to prevent inadvertently draining the IC cubicle pools in the event of an ICC piping failure below the ICS pools. The back flow preventers are installed at the highest elevation of the ICC piping to function as anti-siphon devices. Back flow preventer components are made of Type 304/304L or 316/316L stainless steel. The Back Flow Preventer vent hole is covered by a shield plate located normal to the axis of the vent hole to limit the flow of return processed fluid out of the Back Flow Preventer.

### **Distribution Spargers**

The return water distribution spargers are installed in the IC cubicles near the ICs. The spargers are installed at or near the bottom of the Isolation Condenser cubicles in a manner that promotes bulk flow of the water in the cubicle, particularly with respect to the Isolation Condenser HX. The distribution spargers are fabricated using seamless Type 304/304L or 316/316L stainless steel pipe. The distribution spargers include a flanged end connector to facilitate removal as necessary for maintenance, repair, and replacement.

### **ICC Piping**

Piping in contact with ICS pool water is made of Type 304/304L or 316/316L seamless, stainless steel. ICC piping in contact with PCW on the cold side of the HXs is made of seamless Grade B carbon steel. The ICC piping is designed to comply with ASME B31.1 (Reference 9A.2.6-3) requirements. Drainage taps are provided at low points in the ICC piping to allow the pipes to be drained to facilitate maintenance or repair activities. Vents are provided at the high points in the ICC piping to allow air to be purged from the ICC piping when filling with pool water and to allow

the pool return pipes to be drained to facilitate maintenance or repair activities. Structural supports for piping systems and components inside the ICS Pool Compartment are in accordance with the rules and requirements of ASME BPVC-III NF, Subsection NF, "Supports" (Reference 9A.2.6-4).

### **ICC Valves**

Valve surfaces in contact with ICS pool water are made of Type 304/304L or 316/316L stainless steel. The ICC valves are designed to comply with ASME B31.1 and ASME B16.34 (Reference 9A.2.6-5) requirements. Valve internals in the ICC piping in contact with PCW is made of Type 304/304L or 316/316L stainless steel to maintain leak-tightness. Valves are provided with flanged inlet and outlet connections to facilitate removal as necessary for maintenance, testing, and replacement. The ICS pool suction isolation valve and pool return isolation valves are equipped with manual override and handwheel.

### **Instrument Requirements**

ICC instrument housings and internals in contact with ICS Pool Water are made of Type 304/304L or 316/316L stainless steel. Where possible, instrument housings are provided with flanged connections to facilitate removal as necessary for maintenance, testing, and replacement.

#### **9A.2.6.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, and corrodents through material chemistry, heat treatment, contamination, and material processes controls.

Components and surfaces that come in contact with ICS pool water are manufactured using corrosion-resistant material. Use of corrosion-resistant material reduces iron particulate introduced into the system and Flow Accelerated Corrosion (FAC).

#### **9A.2.6.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.2.6-1 for ICC interfaces with other equipment or systems.

#### **9A.2.6.6 System and Equipment Operation**

The ICC provides active cooling and purification of the ICS pools when it is operating.

The ICC cooling function is required when the ICs are in standby mode due to constant thermal leakage caused by a relatively small amount of condensing steam in the IC during normal reactor operations. The Adjustable Speed Drives allow the ICC to operate at a lower system flow rate during normal operations when the ICs are in standby mode for improved overall plant economy.

The ICC purification function is provided by directing ICC flow through the ICC Demineralizer, which is the normal flow path during routine system operation. The Demineralizer is bypassed for the following conditions:

1. The Demineralizer is bypassed if the temperature of the process fluid exceeds 53.3 °C to prevent thermal degradation of the demineralizer resins and early desorption of adsorbed contaminants that occurs with aging.
2. The Demineralizer is bypassed during the injection of chemicals into the process fluid to inhibit corrosion and biological growth in the ICS Pool Cooling and Cleanup System equipment, piping, and IC pools to prevent chemical degradation of the demineralizer resins.



#### **9A.2.6.6.1 Normal Operation**

Refer to Table 9A.2.6-2, "BWRX-300 Reactor and ICC Operating Modes" which provides a summary of the plant operating modes with corresponding ICC operating modes.

##### **Mode 1: Power Operation**

During Mode 1 power operation, the ICC is in service for all system operating modes:

- ICC Mode A2 – Both ICC trains A and B are operating for maximum heat removal from the ICS pools
  - The Demineralizer is bypassed during Mode A2
- ICC Mode A1a – Only ICC Train A is operating to purify ICS pool water and remove heat
  - The Demineralizer is bypassed if the HX effluent temperature exceeds 53.3 °C
- ICC Mode A1b - Only ICC Train B is operating to purify ICS pool water and remove heat
  - The Demineralizer is bypassed if the HX effluent temperature exceeds 53.3 °C
- ICC Mode A2c – Both ICC trains are operating for injection of water treatment chemicals
  - The Demineralizer is bypassed during chemical injection

One ICC pump/HX train can be isolated as needed for maintenance during power operation. Demineralized makeup water is automatically added to maintain ICS pool inventory during normal reactor operations.

Processed water is returned to individual ICS cubicles as required by aligning the ICS pool isolation valves to the desired ICS cubicle pools.

##### **Mode 2: Startup Operation**

During startup operation, the ICC is in service for all system operating modes described for power operation.

##### **Mode 3: Hot Shutdown Operation**

During hot shutdown operation, the ICC is in service for all system operating modes described for power operation.

##### **Mode 4: Stable Shutdown Operation**

During stable shutdown operation, the ICC is in service for all system operating modes described for power operation. In addition, the ICC can be completely removed from service during stable shutdown operation for system maintenance.

##### **Mode 5: Cold Shutdown Operation**

During cold shutdown operation, the ICC is in service for all system operating modes described for power operation. In addition, the ICC can be completely removed from service during cold shutdown operation for system maintenance.

##### **Mode 6: Refueling Operation**

During refueling operation, ICC is in service for all system operating modes described for power operation. In addition, ICC can be completely removed from service during refueling operations for system maintenance.

#### **9A.2.6.6.2 Off-Normal Operations**

ICC does not have any Off-Normal operating modes.

#### **9A.2.6.6.3 System Shutdown**

The ICC can be completely removed from service for system maintenance during the following plant operating modes:

- Mode 4: Stable Shutdown
- Mode 5: Cold Shutdown
- Mode 6: Refueling

#### **9A.2.6.7 Instrumentation and Control**

Non-Safety-Category function instrumentation and controls (Chapter 7) provides control signals to ICC pumps and valves to initiate system operation and direct flow through system piping branches as needed. The Non-Safety function instrumentation and controls also receives data signals from instruments installed in the ICC piping to measure local process flow variables: water level, temperature, pressure, differential pressure, and conductivity. A local instrumentation monitoring panel is provided in the ICC equipment area for maintenance, diagnostic, and functional testing of the ICC and individual components.

Where possible, instrument housings are provided with flanged connections to facilitate removal as necessary for maintenance, testing, and replacement. Instruments are installed such that there is space surrounding the instrument to facilitate ready access to the instrument and the ability to perform maintenance or replacement.

ICC Control and Alarm and Monitoring functions are identified in Table 9A.2.6-3.

#### **9A.2.6.8 Monitoring, Inspection, Testing, and Maintenance**

Testing of the ICC is performed to demonstrate proper system and component functioning. ICC functionality is continuously demonstrated during normal plant operation. Provisions are made for periodic inspection of major components to ensure the capability and integrity of the system. Service platforms are installed to facilitate inspection of the Surge Tank and Return Guard Pipe. The design of the ICC allows for the inspection and testing of components under normal operating conditions.

Material and equipment selection for the ICC is based on a 60-year design life, with appropriate provisions for maintenance and replacement. Components that require replacement prior to the end of the 60-year design life include, but are not limited to, electrical and electronic equipment, gaskets and seals, lubricants, valve disks and internal components such as seats and packing, and bearings.

Isolation valves are provided near the ICC piping penetrations to the ICS Outer Pool A to permit maintenance and repair activities to be performed on pipe section below the ICS pools. Valves are installed with flanged connections to facilitate removal as needed for testing or repair. In addition, valves and other components are installed with plenty of space surrounding the component to facilitate ready access and the ability to perform maintenance, repair, or replacement activities.

Maintenance isolation valves and flanged connections are provided for the pumps and HXs so they can be isolated and removed for repair or replacement as required. Drain taps are provided at low points in piping sections to evacuate process water for maintenance and repair activities.

A manual crossover feature allows for simultaneous maintenance on the pump of one train with the HX on the other train without interrupting normal cooling and cleanup operation. This design feature adds flexibility for maintenance purposes thereby increasing ICC reliability and availability. The design of the ICC provides adequate equipment removal paths and personnel access for maintenance and repair activities and equipment replacement.

Testing is performed to ensure required functional operability is maintained under design conditions. Testing is performed in accordance with plant procedures. Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Section 14.3.

#### **9A.2.6.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.2.6.10 Performance and Safety Evaluation**

Two features have been incorporated into the ICC design to increase system reliability.

1. A crossover pipe with manual isolation valve is installed between the A and B trains pump discharge to allow the system to remain operable in the event Pump A and HX B are simultaneously out of service for maintenance activities, or if Pump B and HX A are simultaneously out of service.
2. Manual overrides with handwheels are specified for the main suction and for the return Air-Operated Block Valves to enable manual operation of these valves in the event of actuator failure.

Safety Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools. The Suction Surge Tank and Return Guard Pipe are categorized as SC1 and Seismic A components and function to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools.

#### **9A.2.6.11 References**

- 9A.2.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.6-2 ANSI/HI 1.3, "Rotodynamic Centrifugal Pumps for Design and Application," Hydraulic Institute.
- 9A.2.6-3 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.2.6-4 ASME BPVC-III NF, "BPVC Section III – Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NF – Supports," American Society of Mechanical Engineers.
- 9A.2.6-5 ASME B16.34, "Valves – Flanged, Threaded, and Welding End," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Control System	<p>Provides control logic for initiation and control of the ICC. Receives data signals from ICC instruments for display on MCR and local panels</p> <p>Input to ICC: Control signals to pumps and valves</p> <p>Output from ICC: Instrumentation data</p>	Remote Multiplexer Unit associated with ICC instrument and control equipment
Process Radiation and Environmental Monitoring System	<p>Provides process sampling for water quality to detect potentially radioactive contaminated Demineralizer resin before discharge to SWM</p> <p>Output from ICC: Process water samples for analysis for water quality and monitoring Demineralizer performance</p>	<p>ICC main Return Pipe Tie-ins</p> <p>ICC Demineralizer</p>
Isolation Condenser System	<p>Provides process fluid from ICS pools. Provides pool temperature and water level conditions via instruments in the ICS Outer Pool at ICC suction interface</p> <p>Input to ICC: Process fluid and inlet conditions (pool temperature and water level)</p> <p>Output from ICC: Processed fluid is returned to the ICS cubicles</p>	<p>ICS Pool Piping Penetrations</p> <p>ICS A, B, and C Temperature Elements</p> <p>ICS Outer Pool TE and Level Transmitter</p>
Control Panel System	<p>Provides control logic for initiating operation and control of the ICC. Receives data signals from ICC instruments for display on MCR and local panel</p> <p>Input to ICC: Control signals to pumps and valves</p> <p>Output to from ICC: Instrumentation data</p>	Remote Multiplexer Unit associated with ICC instrument and control equipment
Solid Waste Management System	<p>Receives discharged demineralizer resin from ICC Demineralizer</p> <p>Output from ICC: Discharge of potentially radioactively contaminated spent resin media from the Demineralizer</p>	<p>ICC Demineralizer</p> <p>Interface is at downstream side of ICC isolation valve</p>

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Plant Pneumatics System	Provides compressed air for ICC Instrumentation and Control (I&C) components, AOVs, and Demineralizer  Input to ICC: Compressed air service for: ICC I&C; Demineralizer resin media redistribution and exchange evolutions	ICC AOVs ICC Demineralizer  Interface for compressed air is at upstream side of ICC isolation valve
Non-Safety Electrical Distribution System	Provides Non-Safety-Category AC electrical power to ICC equipment  Input to ICC: Non-Safety-Category electrical power to equipment electrical loads in ICC	Interface to be located at local breaker box
Heat, Ventilation, and Cooling System	Provides HVAC for environmental temperature control of ICC Equipment Room  Input to ICC: HVAC air flow Output from ICC: Heat load from ICC equipment	Interface at ICC Equipment Room
Reactor Building Structure	Provides space, structural and infrastructure support, and protection for ICC piping and equipment  Input to ICC: ICC equipment is situated within dedicated space provided by the RBS	Interfaces at: (1) ICC Equipment Room, (2) ICS piping penetrations, (3) ICS pools (pipe supports)
Water, Gas, and Chemical Pads	Provides makeup demineralized water for ICS pools during normal reactor operations  Input to ICC: Demineralized water to replace routine pool losses due to evaporation	Interface for makeup water is at upstream side of ICC isolation valve
Water, Gas, and Chemical Pads	Provides demineralized water to ICC Demineralizer. Input to ICC: Demineralized water to facilitate Demineralizer resin media exchange evolutions	Demineralizer Interface for demineralized water is at upstream side of ICC isolation valve

**Table 9A.2.6-2: Isolation Condenser System Pool Cooling and Cleanup System  
Operating Modes**

Mode	Title	Reactor Mode Switch Position	ICC Modes	Description
1	Power Operation (10 – 100% Rated Power)	RUN	A2	Cooling (A & B Trains Operating)
			A1a	Cleanup (A Train Operating)
			A1b	Cleanup (B Train Operating)
			A2c	Chemical Injection
2	Startup	STARTUP or REFUEL <sup>(1)</sup>	A	All ICC Operating Modes
3	Hot Shutdown <sup>(1)</sup>	SHUTDOWN	A	All ICC Operating Modes
4	Stable Shutdown <sup>(1)</sup>	SHUTDOWN	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance
5	Cold Shutdown <sup>(1)</sup>	SHUTDOWN	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance
6	Refueling <sup>(2)</sup>	SHUTDOWN or REFUEL	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance

(1) All RPV head closure bolts fully tensioned

(2) One or more RPV head closure bolts less than fully tensioned

**Table 9A.2.6-3: Isolation Condenser System Pool Cooling and Cleanup  
System Instrumentation**

Instrument	Location	Function
Temperature Element	ICS Cubicle Pools	Control Temperature of ICS Cubicle Pools used to initiate ICC A2 operation
Temperature Element	Suction Surge Tank	Alarm and Monitoring
Level Transmitter	Suction Surge Tank	Control ICS Pool Water Level used to control automatic addition of makeup demineralized water and shutoff pumps if water level is too low
Pressure Transmitter	Pump Suction	Alarm and Monitoring Suction Pressure
Pressure Transmitter	Pump Discharge	Alarm and Monitoring Discharge Pressure
Pressure Differential Transmitter	Pump Discharge	Control Pressure Drop across Flow Element used to control pump speed to adjust Flow Rates in A and B Trains
Flow Element (Orifice Plate)	Pump Discharge	Alarm and Monitoring Flow Rate in A and B Trains
Temperature Element	HX Inlet (Hot Side)	Alarm and Monitoring Temperature of PCW entering HX Hot Side
Pressure Differential Transmitter	HX Inlet and Outlet (Hot Side)	Alarm and Monitoring Pressure Drop across HX Hot Side
Temperature Element	HX Outlet (Hot Side)	Alarm and Monitoring/HX Condition Monitoring
Temperature Element	Main System Return	Control Temperature of HX effluent into Demineralizer
Conductivity Transmitter	Main System Return	Alarm and Monitoring Pre- and post-Demineralizer Conductivity of return fluid
Pressure Differential Transmitter	Demineralizer Inlet and Outlet	Alarm and Monitoring Pressure Drop across Demineralizer
Pressure Differential Transmitter	Return pipes to Individual IC Cubicles	Alarm and Monitoring Pressure Drop across Flow Element
Flow Element (Orifice Plate)	Returns to Individual IC Cubicles	Alarm and Monitoring Flow Rate in Return Piping

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Instrument</b>	<b>Location</b>	<b>Function</b>
Pressure Transmitter	Makeup Water Line	Alarm and Monitoring Pressure of Makeup Demineralized Water
Pressure Differential Transmitter	Makeup Water Line	Alarm and Monitoring Pressure Drop across Flow Element on Makeup Demineralized Water
Flow Element (Orifice Plate)	Makeup Water Line	Alarm and Monitoring Flow Rate on Makeup Demineralized Water
Pressure Differential Transmitter	HX Inlet (Cold Side)	Alarm and Monitoring Pressure Drop across Flow Element on HX Cold Side
Flow Element (Orifice Plate)	HX Inlet (Cold Side)	Alarm and Monitoring Flow Rate on HX Cold Side
Temperature Element	HX Inlet (Cold Side)	Alarm and Monitoring Temperature entering HX Cold Side
Pressure Differential Transmitter	HX Inlet / Outlet (Cold Side)	Alarm and Monitoring Pressure Drop across HX Cold Side
Temperature Element	HX Outlet (Cold Side)	Control Temperature exiting HX Cold Side used to control Demineralizer Bypass and flow of PCW into HX
AOV Actuators	All AOV Actuators	Alarm and Monitoring Valve Position Indication Limit switches to detect current position (open/closed) of AOVs



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

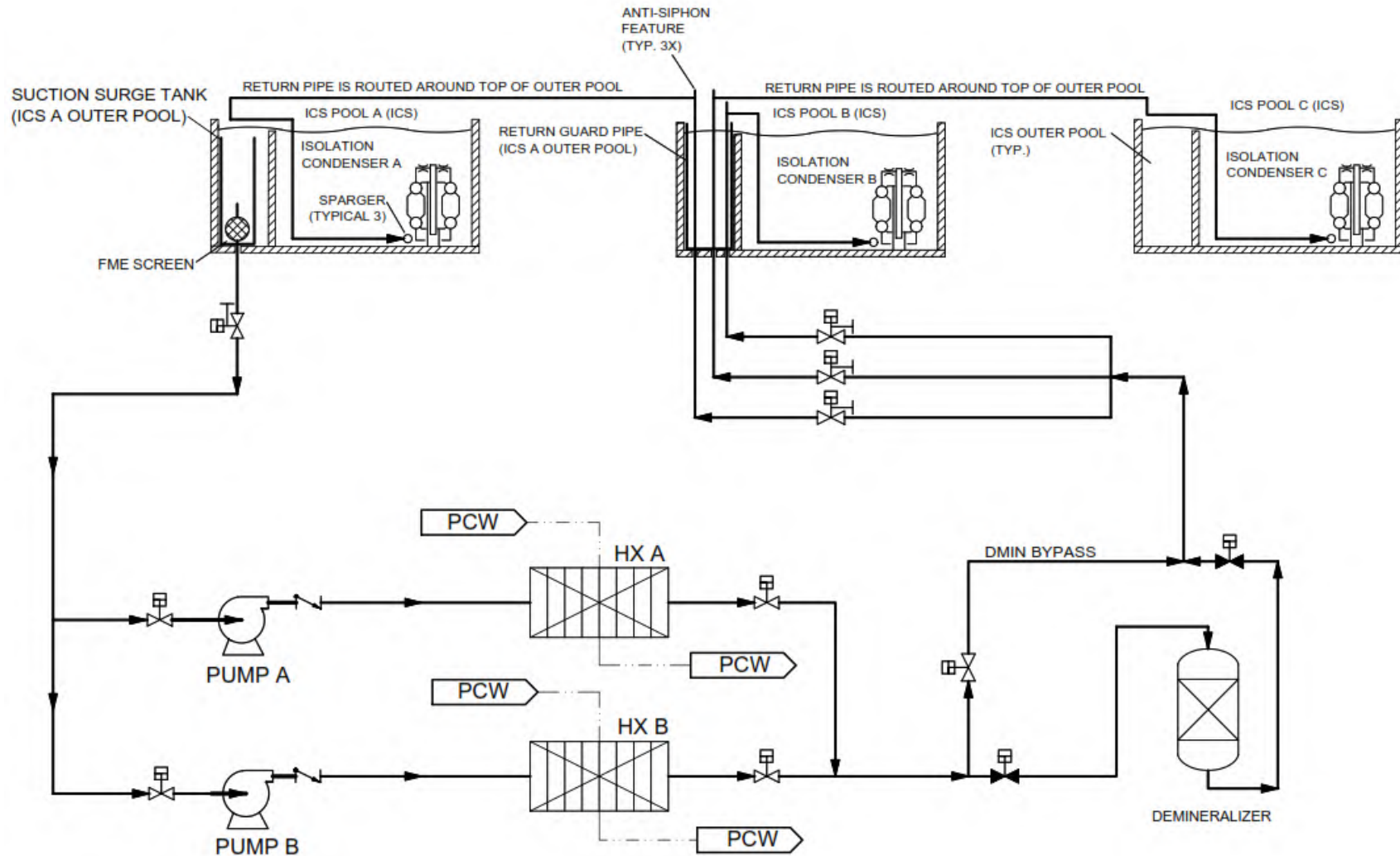


Figure 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System

### **9A.3 Process Auxiliary Systems**

#### **9A.3.1 Process Sampling Systems**

The Process Sampling System is a subsystem of the Process Radiation and Environmental Monitoring System. In addition to the Process Sampling subsystem, the PREMS is comprised of the following subsystems, Process Radiation Monitoring (Chapter 11, Subsection 11.5), Area Radiation Monitoring (Chapter 12, Subsection 12.3.14), and Containment Monitoring (Chapter 12, Subsection 12.3.14).

The Process Sampling subsystem is included in the design of the BWRX-300 based upon the requirements of REGDOC 2.5.2 (Reference 9A.3.1-1) Section 8.13.3 "Radiation Monitoring" as applicable.

The BWRX-300 design does not employ a Post-Accident Sampling System for assessing core damage as per REGDOC 1.1.2 (Reference 9A.3.1-2). In its place is a collection of Post-Accident Monitoring (PAM) inputs utilizing real-time measurements of plant parameters which are deemed more accurate and/or more timely to measure core damage. The sensors and instruments designated for PAM are treated differently by the plant I&C architecture to ensure availability during and after the most severe accident or Beyond Design Basis Accident. Many of the other Post-Accident Sampling System samples were not needed for short severe accident scenario or the information is available from post-accident monitoring instruments such as radiation monitors, (containment, main steam and offgas) temperature and water level as discussed in Reference 9A.3.1-3.

##### **9A.3.1.1 System and Equipment Functions**

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. The Process Sampling subsystem performs the following functions during normal and off-normal conditions.

###### **9A.3.1.1.1 Normal Function (Non-Safety Category)**

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. This process subsequently guides changes to operating parameters. This subsystem is designed to function during all plant operational modes under individual system requirements. The Process Sampling subsystem does not perform or ensure any Safety Class function. Therefore, this system has no Safety Class design basis.

###### **9A.3.1.1.2 Normal Function (Safety Category)**

This subsystem does not perform any Safety-Category functions during normal conditions.

###### **9A.3.1.1.3 Off-Normal Functions (Non-Safety-Category)**

This subsystem does not perform any Non-Safety-Category functions during Off-Normal conditions.

###### **9A.3.1.1.4 Off-Normal Functions (Safety-Category)**

This subsystem does not perform any Safety-Category functions during Off-Normal conditions unless designated a Post-Accident Monitoring function.

#### **9A.3.1.2 Safety Design Bases**

The Process Sampling subsystem does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

#### **9A.3.1.3 Description**

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance.

Process streams not requiring continuous sampling or computer monitoring include provisions for obtaining grab samples. Sample lines are routed to a common sample station, local sample station (or simply local sample taps) depending on plant layout and process fluid.

The Process Sampling subsystem consists of the following: permanently installed sample lines, sampling panels with analyzers and associated sampling equipment, provisions for grab sampling, and permanent shielding. The division of sample lines versus grab samples are assessed during detailed design activities based on the availability of automation in the instrumentation, the need to limit personnel access (ALARA), the frequency of needed measurements, and other considerations. Sampling stations and associated process samples are summarized by location below.

Sample Stations:

- Reactor Water Cleanup System
- Shutdown Cooling System
- Control Rod Drive System
- Fuel Pool Cooling and Cleanup System
- Isolation Condenser System
- Isolation Condenser Pool Cooling and Cleanup System
- Condensate and Feedwater System
- Moisture Separator Reheater System
- Heater Drain and Vent System (Condensate and Feedwater subsystem)
- Nuclear Boiler System Main Steam Lines
- Condensate Filter and Demineralizer System
- Main Turbine Equipment
- Main Condenser
- Liquid Waste Management
- Solid Waste Management

Local Grab Sample Stations:

- Plant Cooling Water System
- Water, Gas, and Chemical Pads (Plant Service Water and Makeup Water are subsystems)
- Chilled Water System

- Circulating Water System
- Equipment and Floor Drain System

Containment isolation function is applied to all mechanical instrument sensing line penetrations of the containment boundary in a manner that provides the highest reliability of maintaining instrument function while limiting potential radioactive release if an instrument line is ruptured outside the containment boundary.

The Process Sampling subsystem monitors plant liquids and gases utilizing components designed and installed in such a way as to minimize interconnection of radioactive and non-radioactive systems. To the extent practicable, Process Sampling subsystem interconnections to non-radioactive systems are limited to purge air, purge water and makeup water for filling loop seals. The designs of these interconnections prevent contamination of the non-radioactive system or process due to leakage, spillage, valving errors, or other operating conditions.

The Process Sampling subsystem is designed to minimize facility and environmental contamination and minimize the generation of radioactive waste. These objectives are supported by the procedures for operation, and are controlled as follows:

- Providing atmospheric purging of the internal portion of air sampling skids as necessary
- Providing the ability for liquid flushing of the internal portions of liquid sampling skids as necessary
- Designing the interior portions of liquid and gaseous sampling chambers to minimize the plate out of radioactive material
- Designing sample extraction points such that they minimize the potential for spillage and contamination of adjacent areas
- Minimizing the generation of liquid radioactive waste by minimizing the amount of a sample that needs to be extracted, consistent with laboratory and sensitivity requirements

#### **9A.3.1.3.1 Component Description**

The Process Sampling subsystem consists of sample lines, sampling panels with analyzers and associated sampling equipment, provisions for grab sampling, and shielding.

#### **9A.3.1.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC (as applicable) through material chemistry, heat treatment, contamination and material processes controls.

#### **9A.3.1.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.3.1.1-1 which provides the description and boundary for each interfacing system.

#### **9A.3.1.6 System and Equipment Operation**

The Process Sampling subsystem operates continuously or in grab sample mode when the interfacing system being monitored is operational.

##### **9A.3.1.6.1 Initial Configuration (Pre-Startup)**

The Process Sampling subsystem operates continuously or in grab sample mode.

#### **9A.3.1.6.2 System Startup**

The Process Sampling subsystem operates continuously or in grab sample mode.

#### **9A.3.1.6.3 Normal Operations**

Sampling equipment operates continuously with capability for grab samples from each process stream.

Root valves for process system sampling points are included in the Process Sampling subsystem. The safety requirements for root valves are determined by the safety requirements for the particular system.

#### **9A.3.1.6.4 Off-Normal Operations**

There are no mitigating system functions tied to the Process Sampling subsystem.

#### **9A.3.1.6.5 System Shutdown**

Process Sampling subsystem is performed in accordance with detailed operation procedures.

#### **9A.3.1.7 Instrumentation and Control**

The Process Sampling subsystem is supported by the SCN Distributed Control and Information System (Chapter 7, Subsection 7.3.4). SCN classified functions are implemented on a hardware/software platform to provide the reliability needed to prevent controller failure from becoming an Anticipated Operational Occurrence (AOO).

#### **9A.3.1.8 Monitoring, Inspection, Testing, and Maintenance**

The Process Sampling subsystem design provides continuous availability regardless of the plant operating mode; therefore, Process Sampling subsystem functionality is continuously demonstrated during all phases of normal, shutdown, and off-normal conditions.

Routine maintenance and cleaning of the sampling subsystem is performed. The design accommodates easy access to sensors, instruments, and panels for maintenance. Additionally, devices for lifting or hoisting the Process Sampling subsystem components is provided to facilitate replacement if required.

Instrumentation is designed to facilitate calibration checks and troubleshooting.

Sampling racks and electronic modules are serviced and maintained in accordance with the operational instructions to ensure reliable operation. Such maintenance includes servicing and replacement of defective components and adjustments as required. Periodic testing or calibration checks is performed as part of the maintenance plan.

#### **9A.3.1.9 Radiological Aspects**

Chapter 12, Subsection 12.3.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.3.1.10 Performance and Safety Evaluation**

The system design provides continuous availability regardless of the plant operating mode. SCN functions are implemented on the Non-Safety Instrumentation and Control System which provides the reliability needed to prevent controller failure from becoming an AOO. The Process Sampling subsystem performs no Safety-Category functions and is not required to prevent or mitigate the consequences of a design basis accident, to shut down the reactor and maintain safe shutdown

conditions, or to maintain the integrity of the reactor coolant pressure boundary. Therefore, a nuclear safety evaluation is not required.

**9A.3.1.11 References**

- 9A.3.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.3.1-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 9A.3.1-3 NEDO-32991-A, "Regulatory Relaxation for BWR Post- Accident Sampling Stations (PASS)," GE Nuclear Energy.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.3.1-1: Process Sampling Subsystem Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Distributed Control and Information Systems	DCIS Non-Safety Class instrumentation communication	Instrumentation communicates with the SCN DCIS system
Nuclear Boiler System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Isolation Condenser System	Process sampling for chemistry control	Local Grab Sample and/or Routed to Sample Station
Control Rod Drive System	Process sampling for chemistry control	Routed to sample station
ICS Pool Cooling and Cleanup System	Process sampling for radiological analysis, chemistry control and demineralizer performance monitoring	Routed to Sample Station
Shut Down Cooling System	Process sampling for chemistry control	Routed to Sample Station
Reactor Water Cleanup System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Fuel Pool Cooling and Cleanup System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Liquid Waste Management System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Solid Waste Management System	Process sampling for radiological analysis, chemistry control, and demineralizer performance monitoring	Routed to Sample Station
Condensate and Feedwater System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Condensate Filters and Demineralizers System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Main Turbine Equipment	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Moisture Separator Reheater System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Main Condenser and Auxiliaries	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Circulating Water System	Process sampling for chemistry control	Local grab sampling
Chilled Water Equipment	Process sampling for chemistry control	Local grab sampling

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Plant Cooling Water System	Process sampling for chemistry control	Local grab sampling
Equipment and Floor Drain System	Process sampling for radiological analysis and chemistry control	Local grab sampling
Water, Gas, and Chemicals Pads	Process sampling for Demineralized Water Tank chemistry control	Local grab sampling



## **9A.4 Air and Gas System**

### **9A.4.1 Plant Pneumatic System**

The Plant Pneumatics System provides a continuous supply of compressed air for the majority of the plant air demands. In addition, PPS supplies oil-free air to service boxes for use of portable breathing air filtration systems. The PPS also distributes nitrogen to users inside containment.

#### **9A.4.1.1 System and Equipment Functions**

##### **Normal Functions (Non-Safety-Category)**

PPS provides the following Non-Safety-Category functions:

1. The PPS provides dry, oil-free, filtered compressed air for valve actuators, instrument control functions, air operated tools, and miscellaneous equipment/services outside of containment in Mode 1, 2, 3, 4, 5, and 6.
2. The PPS provides the compressed air for tank sparging, filter/demineralizer backwashing, air-operated tools and other services requiring air of lower quality in Mode 1, 2, 3, 4, 5, and 6.
3. The PPS provides the distribution of gaseous nitrogen, supplied by the Containment Inerting System (CIS) (Subsection 9A.4.2), to primary containment for valve actuators and other primary containment users in Mode 1, 2, 3, 4, and 5.
4. The PPS supplies instrument-quality air to nitrogen consumers when primary containment is open for personal access (e.g., major outages).
5. The PPS supplies oil-free air to service boxes for breathable air.

##### **Normal Functions (Safety-Category)**

The PPS provides dry, oil-free, filtered compressed air for Safety-Category valve actuators and instrument control functions outside of containment (Safety Class 3).

The PPS provides the distribution of gaseous nitrogen, supplied by CIS, to Primary Containment for Safety-Category valves, controls, instrumentation, and other Primary Containment users (Safety Class 3).

While these functional requirements are categorized as SC3, not all associated air demands are expected to be SC3.

##### **Off-Normal Functions (Non-Safety-Category)**

The power supply to the PPS automatically switches to standby AC power during a LOOP.

##### **Off-Normal Functions (Safety-Category)**

As part of Defense Line 3 the PPS provides containment isolation valves on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Subsection 7.3.1).

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.4.1-1). Refer to Chapter 6, Subsection 6.3.4 for information pertaining to containment isolation. In addition, the design of the PPS meets the requirements specified in CNSC REGDOC-2.5.2 Section 7.1 as related to ensuring that the Non-Safety-Category function portions of the PPS are isolated from the Safety-Category function portions of the PPS.

#### **9A.4.1.2 Safety Design Bases**

The PPS is Safety Class 3, providing integral support functions for DL2 functions:

As part of the PPS Safety-Category 3 functions the PPS provides dry, oil-free, filtered, compressed air for valve actuators, Safety-Category instrument control functions, and general instrumentation services outside of the Steel-plate Composite Containment Vessel (SCCV). The PPS provides the distribution of gaseous nitrogen, supplied by the CIS, to SCCV for valve actuators and other SCCV users.

Containment Isolation Valves are provided on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.

#### **9A.4.1.3 Description**

Figure 9A.4.1-1 depicts the PPS.

The PPS consists of two 100% air compressors, two 100% capacity dryer trains, two service air receivers operating in parallel upstream of the dryers, and two instrument air receivers operating in parallel downstream of the dryers. The PPS provides service and instrument air for the majority of plant air needs.

Each compressor takes suction through an air intake filter/silencer which brings air from outside the TB. The air inlet is fitted with rain hood and bird screen. The lead compressor supplies air to both service air receivers, which will flow through the lead dryer train, to both instrument air receivers.

Distribution piping in the plant is by ring headers which allows for flexibility in distribution locations, maintenance, and ease of accommodating future requirements. The service air receivers distribute air to the service air header demands. The service air subsystem also connects to a separate utility containment penetration to provide service air to the service boxes inside containment. These service boxes are used to supply air to portable breathing air filtration systems to provide breathing air hookups to personnel entering containment when the atmosphere is not yet fully habitable or for areas where high levels of airborne contaminants cannot be eliminated efficiently by the HVAC.

The instrument air receivers will distribute air to the instrument air header demands. The instrument air subsystem also includes piping penetrating containment, connecting the CIS to pneumatic valves and other compressed gas users inside containment. During plant outages, the PPS supplies the pneumatic valves and other compressed gas users inside containment with instrument-quality air while the nitrogen supply from the CIS is isolated from inside containment.

During normal operation, the air supply is isolated from the primary containment vessel via double block and bleed valve configuration so that nitrogen from the CIS can be supplied to nitrogen users inside containment using the PPS primary containment piping penetrations. The block and bleed valves on the air supply prevent air from entering containment during normal operation and posing an explosion and fire risk. During plant shutdown, the nitrogen supply is isolated from the primary containment vessel via double block and bleed valve configuration, so that in the case of a leak in the PPS supply piping to primary containment, the primary containment does not turn into a confined space or pose an asphyxiation risk to plant personnel.

PPS is designed to meet the applicable requirements of Instrument Society of America 7.0.01 (Reference 9A.4.1-2).

#### **9A.4.1.3.1 Component Description**

The following paragraphs provide information that pertains to the major equipment items in the PPS system.

##### **Air Compressors**

The two (2) compressors are oil-free, rotary screw, water-cooled, electric motor driven compressors. Each compressor can supply 100% of the system's continuous air requirements. Noise reduction methods are incorporated to ensure that noise levels in habitable areas near the compressors are within the guidelines for industrial environments. The air compressor units are powered from the two (2) separate electrical busses. The service air piping includes a connection for adding an additional compressor. These compressors are commercially available skid packages.

##### **Air Receivers**

The system includes two service air receivers in parallel and two instrument air receivers in parallel. The design capacity of the receivers includes adequate reserve capacity to provide at least 10 minutes of operating time following a trip with no air compressors in operation before the low pressure set point alarms are sounded in the control room.

The receivers are ASME VIII designed and stamped.

A pressure-relief safety valve is installed on each receiver.

##### **Dryer/Filter Trains**

The PPS utilizes two (2) 100% dryer trains. The dryers are regenerative desiccant air dryers.

The dryer trains include coalescing pre-filters and after-filters. The 100% capacity pre-filters are included to remove water and oil droplets, rust, dust, and other solid objects suspended in the air. The 100% capacity after-filters are included to remove particles of desiccant that may be carried away from the dryer. During normal operation, one dryer/filter train is continuously operating, and the other train is in standby. If both compressors are operating, both dryers are placed in service, if available.

The dryer filter trains deliver compressed air that meets ISA 7.0.01 (Reference 9A.4.1-2) Quality Standards for Instrument Air.

##### **Piping and Valves**

All PPS piping is welded stainless steel material

The piping, other than that required for containment isolation, is designed to meet the requirements of ASME B31.1.

To ensure containment integrity at the Steel Composite Containment Vessel penetrations, the containment piping and isolation valves are designed to Seismic Category A and B, respectively; ASME BPVC, Section III, Division 1-Subsection NCD; and Quality Group B requirements.

PPS containment isolation piping includes an AOV outside of containment and a check valve inside of containment. During a Design Basis Accident (DBA), if the AOV fails to close leakage out from inside containment is stopped by the air system if an air compressor is operating or by the check valve if an air compressor is not operating.

#### **9A.4.1.4 Materials**

The PPS is designed utilizing materials that ensure that the functional requirements of the system are achieved. Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

#### **9A.4.1.5 Interfaces with Other Equipment or Systems**

Table 9A.4.1-1 identifies PPS interfaces with other BWRX-300 Systems.

#### **9A.4.1.6 System and Equipment Operation**

The PPS operates in the normal plant environment and provides Non-Safety and SC3 functions. With the exception of the containment isolation valves the PPS is not required to operate after a design basis accident.

The system is designed for continuous operation during all modes of plant operation:

##### **Normal Operational Concept**

During normal operation, one 100% capacity air compressor operates to supply both service air receivers, one dryer train, both instrument air receivers, and both instrument and service air system distribution piping.

A compressor is chosen as the lead, continuously operating compressor. The other compressor serves as standby.

The standby compressor automatically starts upon trip of the lead compressor, or when the air pressure in the system drops below the predetermined pressure set point. The assignment for lead and standby air compressors is switched periodically by operators in the MCR.

One 100% dryer/filter train operates, and one serves as standby. Automatic start of the standby dryer skid occurs when the differential pressure across a dryer filter, discharge pressure or the dewpoint reach predetermined setpoints. The assignment for lead and standby dryers is switched periodically by operators in the MCR.

##### **Off-Normal Operational Concept**

The PPS containment isolation valves and associated piping are designed to be functional after a safe shutdown earthquake.

When a LOOP is detected, the operating compressor and dryer both trip and restart when power is received from the backup diesel generators.

#### **9A.4.1.7 Instrumentation and Control**

##### **9A.4.1.7.1 Instrumentation**

The PPS contains instrumentation to remotely monitor (indicate and alarm) and control all equipment performance, control valve status, and necessary system parameters, such as pressure, temperature, and moisture content of the compressed air in the system. Data signals from the PPS instruments and equipment are displayed on the Main Control Room (MCR) and local instrumentation panels. Local control panels for the compressors and dryers are used to operate during testing, calibration, and changing standby compressor and dryer units.

Instrumentation associated with one or more control functions are redundant. For additional information refer to Chapter 7, Subsection 7.3 "Distributed Control and Information System

Functions". This includes instruments such as vendor package (skid) instrumentation that send equipment trip signals or standby equipment start signals upon detecting specified abnormal operating conditions. Additionally, service and instrument air header pressure sensors that provide the main control parameters for the PPS are redundant, described in the Controls section below.

#### **9A.4.1.7.2 Controls**

The PPS is automatically controlled from the plant Distributed Control and Information System (DCIS). The start and stop of the PPS compressors and dryers are automatically or manually operated from the MCR. The instrumentation and control architecture in the PPS is designed to keep the instrument air available in all modes of operation. All automatic or remotely actuated valves in the PPS fail in the position most likely to keep the system operational and controllable with the exception of the PPS containment isolation valves.

The pressure in the service air header is the main control parameter in the system. The operating compressor loads/unloads based on pressure readings from the redundant pressure sensors on the service air header. Upon a loss of both compressors, or at a low pressure setpoint, the normally open service air isolation valve closes to shed the service air system load and conserve air for the instrument air subsystem to maintain control of the plant. Upon a loss of both dryers, or a low-low pressure indication from the redundant instrument air header pressure sensors, the dryer bypass line opens to allow the flow of service air to the instrument air users.

#### **9A.4.1.8 Monitoring, Inspection, Testing, and Maintenance**

All major components have manual isolation valves to permit maintenance activities without having to shut down the PPS. The maintenance boundary between the isolation valves is designed to permit the simultaneous maintenance of the maximum number of components as possible without impacting the ability of the system to remain in service.

PPS piping and valves for the containment penetrations are tested in accordance with 10CFR50, Appendix J and CNSC REGDOC-2.5.2. Test and vent connections are provided at the containment isolation valves and are used to verify that the valves meet the local leak rate limits.

The PPS outboard containment isolation valve is tested to ensure operational integrity by manual operation of the valve. The inboard containment check valve is periodically tested to ensure valve operability.

All PPS equipment is arranged to provide adequate pull space and clearance for personnel and testing equipment access to facilitate, testing, repair, and maintenance.

Breathing air is sampled in accordance with CSA Z180.1:19 to ensure the CSA breathing air quality standards are met.

Instrument air and system functionality is tested in accordance with ISA 7.0.01 (Reference 9A.4.1-2) and associated guidelines to ensure the instrument air quality standards are met.

#### **9A.4.1.9 Radiological Aspects**

Chapter 12, Subsection 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.4.1.10 Performance and Safety Evaluation**

The SC1 Safety-Category function performed by the PPS is containment isolation. Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.4.1-

1). The PPS CIVs are designed to maintain the leak tightness of the containment in the event of an accident and prevent radioactive releases to the environment that exceed prescribed limits. The PPS incorporates features that ensure its operation over the full range of normal plant operations.

**9A.4.1.11 References**

9A.4.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

9A.4.1-2 ANSI/ISA-7.0.01, "Quality Standard for Instrument Air," Instrument Society of America.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.4.1-1: Plant Pneumatics System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	NBS Equipment
Safety Class 1 Instrumentation and Controls System	Safety Class 1 Instrumentation and Controls System de-energizes the Containment Isolation Valves (CIVs) solenoids which release the air inside the actuators.	PPS Equipment
Safety Class 2 and 3 Instrumentation and Controls System	Safety Class 2 and 3 Instrumentation and Controls System provide signals to SC3 instrumentation and controls in PPS.	PPS Equipment
Non- Safety Instrumentation and Control System	Receive/provide signals to non-safety category instrumentation and controls in PPS.	PPS Equipment
Process Radiation and Environmental Monitoring System	PPS provides control and instrument air to AOVs and equipment/components as necessary. Provides periodic sampling of breathing air quality and instrument air quality.	PREMS AOVs
Isolation Condenser System	PPS provides control and instrument air to AOVs and equipment/components as necessary. including ICS isolation accumulators.	ICS AOVs
Refueling and Servicing Equipment System (RES)	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At RES Equipment
Boron Injection System (BIS)	PPS provides Service Air for periodic storage tank mixing through the in-tank air sparger.	At BIS Equipment
Control Rod Drive System/High Pressure Injection	PPS provides instrument air to the HCU scram valves, and CRD flow control AOVs and Alternate Rod Insertion valves.	CRD Equipment
Isolation Condenser System Pool Cooling and Cleanup System	PPS provides instrument air for ICC I&C components, AOVs, and DMIN to facilitate demineralizer resin conditioning and exchange evolutions.	ICC AOVs and DMIN
Shut Down Cooling System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At SDC Equipment
Reactor Water Cleanup System	PPS provides control and instrument air or nitrogen to AOVs.	At CUW AOVs

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Fuel Pools Cooling and Cleanup	PPS provides control air for mixing of demineralizer resin and instrument-quality air for controlling the FPC AOVs.	At FPC AOVs and demineralizer resin tank
Liquid Waste Management System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At LWM AOVs
Solid Waste Management System	PPS instrument air to AOVs and dewatering pumps.	At SWM AOVs and SWM dewatering pumps
Offgas System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At OGS AOVs
Condensate and Feedwater Heating System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CFS AOVs
Condensate Filters and Demineralizers System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CFD AOVs
Main Turbine Equipment	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MTE AOVs
Moisture Separator Reheater System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MSR AOVs
Main Condenser and Auxiliaries	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MCA AOVs
Circulating Water System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CWS AOVs
Chilled Water Equipment	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CWE AOVs
Plant Cooling Water System	PPS provides control and instrument air to AOVs and equipment/components as necessary. PCW provides cooling water to the air compressors.	At PCW AOVs and compressor cooler isolation valves



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Hydrogen Water Chemistry	PPS provides control and instrument air to AOVs and equipment/components as necessary. The PPS also provides air supply for air injection to the OGS.	At Hydrogen Water Chemistry AOVs
Safety Class 2 and 3 Electrical Distribution System	Safety Class 2 and 3 Electrical Distribution System provides electrical power to PPS.	At PPS equipment
Primary Containment System (CON)	PPS provides CIVs on piping penetrating primary containment, and functions as an extension of the containment boundary.	At containment penetrations
Containment Inerting System	PPS provides control and instrument air to AOVs and equipment/components as necessary. CIS provides nitrogen to components inside of containment through the PPS.	At CIS AOVs
Fire Protection System (FPS)	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At FPS AOV
Equipment and Floor Drain System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At EFS AOVs
Reactor Building Structure	PPS provides hookups for pneumatic tools throughout the RBS.	At Building Structure
Turbine Building Structure (TBS)	PPS provides hookups for pneumatic tools throughout the TBS.	At Building Structure
Radwaste Building (RWB)	PPS provides hookups for pneumatic tools throughout the RWB structure.	At Building Structure

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

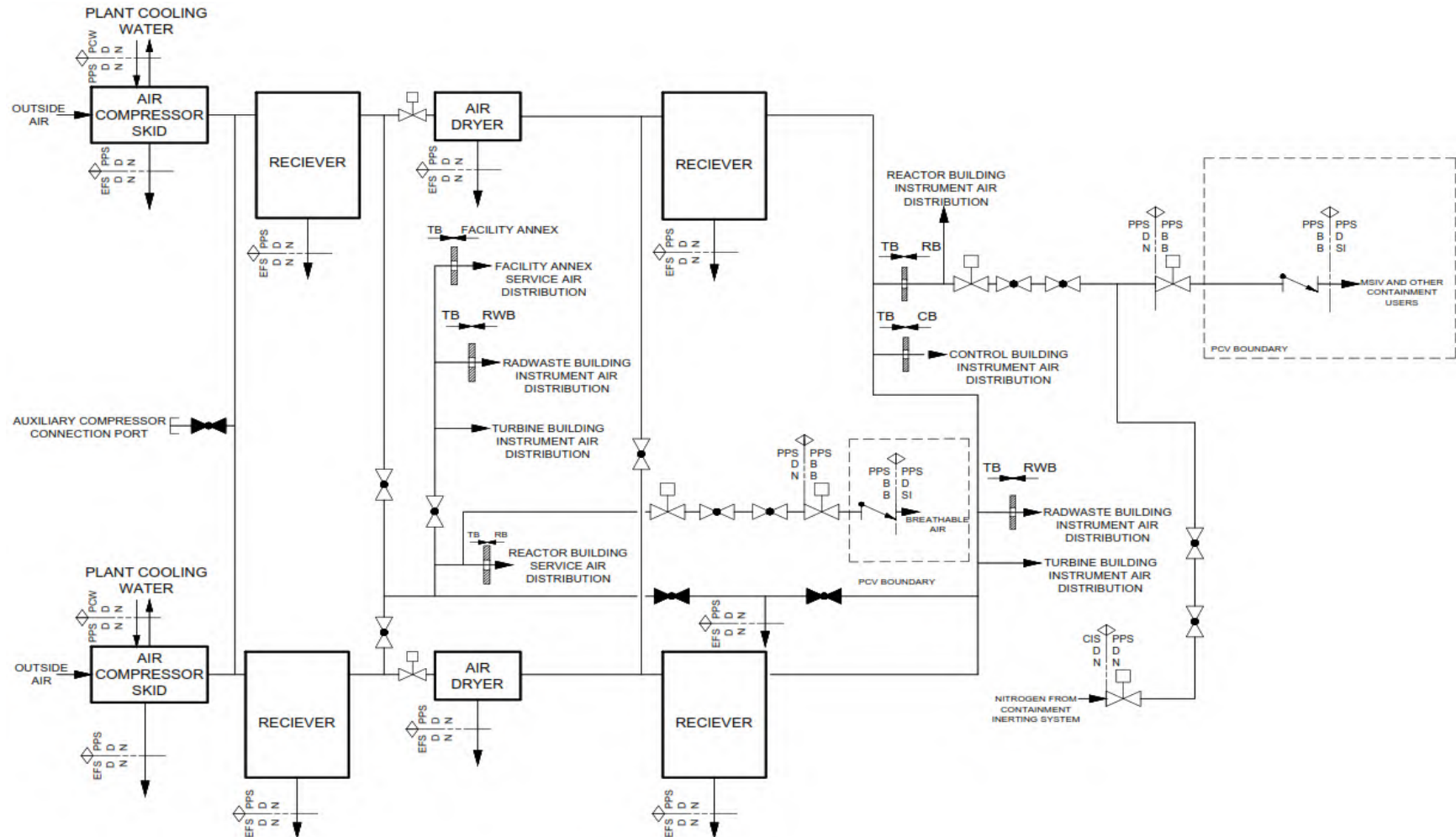


Figure 9A.4.1-1: Plant Pneumatic System

#### **9A.4.2 Containment Inerting System**

The Containment Inerting System is designed to establish and maintain an inert atmosphere (nitrogen) within the Steel-plate Composite Containment Vessel (SCCV). An inert atmosphere is maintained in all operating modes except during plant shutdown for refueling, equipment maintenance, and during limited periods of time to permit access for inspection at low reactor power. Prior to reactor shutdown the CIS can de-inert the SCCV allowing for safe personnel access without the need of a breathing apparatus. The principal objective of the CIS is to preclude the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside SCCV.

The containment boundary is classified as Defense Line 3, Safety Class 1 (DL3/SC1) system.

The containment overpressure vent flow path is Defense Line 4b, Safety Class 3 (DL4b/SC3).

The containment supply and exhaust flow paths are (TBD).

##### **9A.4.2.1 System and Equipment Functions**

The BWRX-300 CIS provides the following functions during Normal and Off-Normal conditions.

###### **9A.4.2.1.1 Normal Functions (Non-Safety Category)**

Establish and maintain an inert atmosphere within containment for prevention of hydrogen combustion during post-accident events that may result in fuel damage. There is no significant hydrogen generation during normal operations.

Maintain a positive pressure within containment to prevent the infiltration of non-inert air during all normal plant operating modes, except plant shutdown

Establish a breathable de-inert atmosphere within containment during plant shutdown for refueling and maintenance

###### **9A.4.2.1.2 Normal Functions (Safety Category)**

The CIS penetrates containment and therefore must ensure containment boundary integrity and isolation capability.

###### **9A.4.2.1.3 Off-Normal Functions (Non-Safety Category)**

Maintain containment oxygen concentration below the maximum permissible limit during abnormal and accident conditions to ensure an inert atmosphere.

###### **9A.4.2.1.4 Off-Normal Functions (Safety Category)**

Protect against containment overpressurization, which could occur during degraded core conditions after a severe accident

The CIVs and associated piping and penetrations are designed in accordance with the rules and requirements of ASME BPVC Code, Section III, Division 1, Subsection NE-Class MC Components, and Subsection NCD, Class 2 Components, in accordance with their quality group classification. The containment pipe penetrations are Seismic Category A and the CIVs are Seismic Category B.

The design and application of the CIS system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.10 (Reference 9A.4.2-1) as related to preventing deflagration or detonation that could jeopardize the integrity or leak tightness of the containment. In addition, containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6.

#### **9A.4.2.2 Safety Design Bases**

The Safety Design Bases of the CIS are as follows:

1. As part of Defense Line 3, the Containment Inerting System provides CIVs on piping that penetrates the containment boundary.
2. As part of Defense Line 3, the Containment Inerting System CIVs are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
3. To protect against containment overpressurization a containment overpressure vent flow path is provided. The containment overpressure vent flow path is Defense Line 4b, Safety Class 3 (DL4b/SC3).

#### **9A.4.2.3 Description**

Figure 9A.4.2-1 depicts the CIS. Components of CIS are located both inside and outside the reactor building. The nitrogen storage equipment, including the liquid nitrogen storage tank, nitrogen vaporizer, and heater are located in the yard.

The CIS provides the capability to supply containment with nitrogen to ensure that a noncombustible atmosphere is maintained following an accident that results in 100% reaction between the fuel cladding and water.

The CIS is capable of reducing the containment oxygen concentrations from atmospheric conditions to less than 3.5% by volume in less than 4 hours.

CIS is also capable of de-inerting the inerted containment to above 19.5% oxygen by volume within 4 hours to allow personnel safe access within containment without the use of a breathing apparatus.

The CIS consists of containment supply, containment exhaust, and containment overpressure venting flowpaths.

##### **Containment Supply Flowpath**

The containment supply flow path is used to inject nitrogen into containment for inerting (large nitrogen flow) and makeup (small nitrogen flow), inject breathable air into containment for de-inerting, and supply nitrogen to the PPS (Subsection 9A.4.1) to support actuation of air operated devices located inside containment. The containment supply flow path consists of a pressurized liquid nitrogen storage tank, vaporizer to convert the liquid nitrogen to a gaseous phase, electrical heater, piping, valves, instruments, and controls. Most of the components are located within the Reactor Building except for the storage tank, vaporizer, heater, associated valves, and piping, which are located in the yard. The containment supply flow path terminates in lower containment.

##### **Containment Exhaust Flowpath**

The containment exhaust flow path is available to be used to discharge the containment atmosphere during the inerting, makeup, and de-inerting process. The containment exhaust flow path consists of piping, valves, instruments, controls, and a tee to the containment overpressure vent flow path. All of the components are located within the Reactor Building. The exhaust flow path starts in upper containment on the opposite side of containment from the supply injection point and connects to the Heating Ventilation and Cooling System (Subsection 9A.5) before discharging to the vent stack.

### **Containment Overpressure Vent Flowpath**

The containment overpressure vent flow path is used in case of a severe accident where containment failure by overpressure is threatened. The containment overpressure vent flow path consists of a rupture disc, a locked closed manual bypass valve, pressure indicator, isolation valve, check valve, sparger, and associated piping. The overpressure vent flow path connects to the containment exhaust flow path and terminates in the Reactor Equipment Pool. By relieving pressure into the Reactor Equipment Pool, water scrubbing is expected to occur, thereby reducing the contaminants. Additionally, the Reactor Equipment Pool is located inside the Reactor Building such that further holdup of any potential released contaminants is provided.

The rupture disk is designed in accordance with the rules and requirements of ASME BPVC, Section III, Class 2 Components. The rupture disk is designated Quality Group A. The containment pipe penetrations are Seismic Category A. The CIVs and rupture disk are Seismic Category B.

The over-pressurization vent line and associated sparger are Safety Class 3.

Environmental Qualification of Safety-Class SSC's is provided as applicable as discussed in Chapter 3, Subsection 3.9.

#### **9A.4.2.3.1 Component Description**

The following information is provided relative to the major equipment items in the ICC system.

### **Liquid Nitrogen Storage Tank**

An insulated liquid nitrogen storage tank provides nitrogen supply to support CIS operations. The CIS liquid nitrogen storage tank is sized to provide enough nitrogen to support two consecutive nitrogen purges of containment. The tank includes level and pressure indicators, and the bottom of the tank is sloped to facilitate draining. Pressure is maintained automatically in the tank during nitrogen discharge by a circuit with an ambient heat exchanger fed by a pressure control valve. The vapor space can be manually vented to control tank internal pressure. The tank is fabricated of material suitable for cryogenic service in order to withstand low liquid nitrogen temperatures. The tank is equipped with quick disconnect/connect or equivalent type of connection to support refilling.

### **Vaporizer**

The nitrogen vaporizers are used to convert the liquid nitrogen from the liquid nitrogen storage tank to the gaseous phase. The CIS nitrogen vaporizers are sized to provide at least 2.5 times the containment free volume of nitrogen within the 4-hour inerting period. The CIS nitrogen vaporizers are fabricated of materials suitable for cryogenic service.

### **Heater**

Vaporization of nitrogen for inerting and makeup is provided by an ambient air heat exchanger with auxiliary electrical heating. The CIS nitrogen heaters are designed to heat the nitrogen makeup to a temperature greater than 10 °C. The CIS nitrogen heaters are fabricated of materials suitable for cryogenic service.

### **Rupture Disk**

The rupture disk is part of the containment over-pressurization vent flowpath and provides passive overpressurization protection for containment. The CIS rupture disk burst pressure tolerance at the specified disk temperature does not exceed 5% of marked burst pressure.

#### **9A.4.2.4 Materials**

CIS Structures, Systems and Components are fabricated of materials suitable for cryogenic service as required.

#### **9A.4.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.4.2-2 for CIS interfaces.

#### **9A.4.2.6 System and Equipment Operation**

Table 9A.4.2-1, "BWRX-300 Operating Modes" provides a summary of the plant operating modes relative to CIS operations.

##### **Initial Configuration (Pre-Startup)**

The initial configuration of the CIS is set to inerting Mode with the CIVs closed. The CIS is in standby readiness condition prior to reactor startup.

##### **System Startup**

During System Startup (Mode 2), the SCCV hatches are closed and the CIS is in-service. Coming out of shutdown where the SCCV was opened or any condition where the oxygen concentration is above the acceptable level, the CIS is set to inerting mode. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower SCCV while exhausting from the upper SCCV until the correct oxygen concentration is reached. The nitrogen is mixed with the SCCV atmosphere by the Containment Cooling System fans. Once the targeted oxygen concentration and pressure levels are reached, the CIS Containment Isolation Valves close.

Once inerting is complete, the CIS can be set to makeup mode to maintain the required oxygen concentration and a slightly positive pressure inside the SCCV to preclude oxygen infiltration from the Reactor Building. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower SCCV. During SCCV atmospheric heating conditions during plant startup, the exhaust line is opened to relieve excess SCCV pressure.

##### **Normal Operations**

During Power Operation (Mode 1), the containment vessel hatches are closed, and the CIS is in-service. The CIS makeup mode is activated when the SCCV pressure drops below the desired setpoint and/or the oxygen concentration rises above the desired setpoint. Positive pressure is needed in the SCCV to preclude oxygen infiltration from containment leakage. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into lower SCCV. During the maximum containment atmospheric heating conditions, the containment exhaust line is opened to maintain SCCV internal pressure.

During Refueling (Mode 6), the CIS is idle. The containment vessel is open and the CIS is rendered ineffective.

##### **Off-Normal Operations**

In the event of a severe accident, resulting in SCCV flooding and elevated internal SCCV pressure, the rupture disc within the containment overpressure vent flowpath will passively open to vent the internal SCCV pressure. The rupture disc is passive requiring no operator action to open. However, an operator controlled manual bypass valve around the rupture disc can be used as an alternate method of venting.

The SCCV internal pressure is monitored in the main control room or by the local pressure indicator to support manual operation. The overall SCCV pressure decreases as venting continues. Once the SCCV internal pressure has been reduced to a safe level and normal containment heat removal capability has been regained, the containment overpressure protection air-operated isolation valve is closed to reestablish the SCCV pressure boundary. Radiological conditions associated with severe accidents are presented in Chapter 15, Subsection 15.3.

### **System Shutdown**

During System Shutdown (Modes 3, 4, and 5), if the containment vessel hatches are required to be opened to support personnel entry, the CIS is set to the de-inerting mode. De-inerting mode is used to achieve a safe, breathable atmosphere once the reactor reaches a setpoint temperature. In de-inerting mode, the CIS injects breathable air into the lower SCCV while exhausting from the upper SCCV until the correct oxygen concentration is reached. The breathable air is mixed with the SCCV atmosphere by the Containment Cooling System fans. Once the targeted oxygen concentration is reached for a breathable atmosphere, the containment vessel hatches may be opened for personnel entry. Opening the containment vessel hatches will render the CIS ineffective, and thus it will be idle.

If the containment vessel is not required to be opened during System Shutdown (Modes 3, 4, and 5), the CIS may be set to makeup mode to maintain an inerted atmosphere in containment.

Refer to Chapter 1, Subsection 1.8 for a description of the operating modes.

#### **9A.4.2.7 Instrumentation and Control**

The following instrumentation features are included for monitoring CIS operations:

1. Flow transmitters for containment inerting flow indication
2. Temperature elements to monitor the inerting nitrogen supply temperature
3. Pressure transmitters for control of the inerting pressure control valves

Features provided for control of CIS operation include the following:

1. Containment Pressure indication provided by Process and Radiation Monitoring System (Chapter 11, Subsection 11.5)
2. Inerting nitrogen supply temperature and pressure for control of the inerting and makeup process
3. Position indication for all remotely operated valves

The following CIS displays and alarms are provided:

1. Main Control Console Indications:
  - a. Containment inlet flow rate
  - b. Position indication (full open and full closed) for all containment isolation valves
  - c. CIS control valves position indication
  - d. Containment oxygen concentration
  - e. Containment inlet temperature
2. Main Control Room Alarms
  - a. High and low nitrogen discharge temperature

- b. Low nitrogen storage tank liquid level
- c. Low and high containment inlet temperature
- d. CIS valve operation failure
- e. High oxygen levels in containment

#### **9A.4.2.8 Monitoring, Inspection, Testing, and Maintenance**

The following pertains to testing and inspection of CIS components:

1. Periodic testing is performed to demonstrate the ability of CIS to meet design requirements. Each valve is exercised both open and closed and the position indication is verified. Trip and alarm logic signals are also checked. The tests assure correct functioning of all controls, instrumentation, piping and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the pre-operational tests and are used as baseline for measurements in the subsequent operational tests. Testing is performed both during operations and shutdown.
2. During plant operation, CIS valves, instrumentation, wiring and other components outside the containment can be inspected visually at any time.
3. Containment isolation valves are tested for leakage periodically in accordance with the requirements of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Type Test C.
4. The rupture disk is tested and replaced periodically, providing additional confidence of pressure.

Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Subsection 14.3.

Testing is performed to ensure required functional operability is maintained under design conditions. The Non-Safety-Category functions of the CIS are tested in accordance with ASME Code B31.1. The Safety-Category functions of the CIS are tested in accordance with ASME BPVC-III.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

Other maintenance provisions include the following:

1. Isolation valves are provided on both sides of the supply and exhaust control valves to isolate the control valves and allow for maintenance and repair.
2. Pull spaces are provided around valves to allow for removal and maintenance.

#### **9A.4.2.9 Radiological Aspects**

Chapter 12, Section 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.



#### **9A.4.2.10 Performance and Safety Evaluation**

The CIS Safety-Category functions during normal conditions include ensuring containment boundary integrity and isolation capability. During Off-Normal conditions, CIS Safety-Category functions include providing protection against containment overpressurization.

The following features are incorporated into the design of the CIS to enhance reliability of the CIS Safety-Category functions:

1. Periodic system testing, and preventive maintenance.
2. If the containment supply pressure control valve fails, a manual bypass valve can be opened to supply nitrogen to the containment. Pressure relief valves are positioned downstream of the pressure control valve to protect the downstream piping and provide for pressure control during manual bypass of the pressure control valve. The manual bypass function allows for continuous operation of CIS while the pressure control valve is out of service.
3. The CIS containment isolation valves are located outside of the SCCV which removes them from the harsh environment of the containment and protects them from the effects of flood and dynamic effects of pipe breaks. The CIVs are accessible for inspection and testing during reactor operation. Also, the CIVs fail closed on loss of air or control signal.
4. A manual bypass valve around the containment overpressure vent rupture disc can be used as an alternate method of venting. The bypass line is equipped with a local pressure indicator that can be used to support early containment venting during a beyond design basis accident.

The CIS structures, systems, and components that provide Safety-Category functions are located in the Reactor Building which provides adequate protection against natural phenomena ensuring the ability of the CIS to perform its Safety-Category functions.

#### **9A.4.2.11 References**

- 9A.4.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

**Table 9A.4.2-1: BWRX-300 Operational Modes**

<b>Mode</b>	<b>Title</b>	<b>Reactor Mode Switch Position</b>	<b>CIS Modes</b>
1	Power Operation	Run	Inerting or Makeup
2	Startup	Refuel or Startup	Inerting or Makeup
3	Hot Shutdown	Shutdown	De-Inerting or Idle
4	Stable Shutdown	Shutdown	De-Inerting or Idle
5	Cold Shutdown	Shutdown	Idle
6	Refueling	Shutdown or Refueling	Idle

**Table 9A.4.2-2: Containment Inerting System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Instrumentation and Control System	The CIS provides instrument data to the Safety Class 2 and 3 Instrumentation and Control System for system monitoring.	Process Parameter Inputs and component status to CIS System Logics
	The Safety Class 2 & 3 Instrumentation and Control System provides a control signal to the CIS solenoid valves.	Nitrogen Supply Isolation Valve Fresh Air Supply Isolation Valve Primary Exhaust Line Control Valve Makeup Exhaust Line Control Valve
Safety Class 1 Instrumentation and Control System	The CIS provides instrument data to the Safety Class 1 Instrumentation and Control System for system monitoring.	Process and Equipment status of safety related components in the CIS system
	The Safety Class 1 Instrumentation and Control System provides an actuation signal to the CIS containment isolation solenoid valves.	Outboard Supply Line Containment Isolation Valve Inboard Supply Line Containment Isolation Valve Inboard Exhaust Line Containment Isolation Valve Outboard Exhaust Line Containment Isolation Valve Overpressure Vent Line Isolation Valve
Reactor Building Structure	The CIS provides a containment overpressure relief path to the Equipment Pool of the Reactor Building Structure for a beyond design basis accident.	Containment Overpressure Vent Line Sparger
SCCV	The CIS penetrates the SCCV to support containment inerting, de-inerting, and makeup.	Supply Line Containment Mechanical Penetration Exhaust Line Containment Mechanical Penetration

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Heating Ventilation and Cooling System	The CIS exhausts containment air to the Heating Ventilation and Cooling System to support containment inerting and de-inerting.	CIS Exhaust Line to HVS Building Exhaust Line downstream of Exhaust Line Control Valves
	The Heating Ventilation and Cooling System supplies breathable air to the CIS to support containment de-inerting.	Fresh Air Supply Isolation Valve
Plant Pneumatics System	The Plant Pneumatics System supplies compressed air to the CIS to support actuation of pneumatically operated devices.	Nitrogen Supply Isolation Valve Fresh Air Supply Isolation Valve Primary Exhaust Line Control Valve Makeup Exhaust Line Control Valve Outboard Supply Line Containment Isolation Valve Inboard Supply Line Containment Isolation Valve Inboard Exhaust Line Containment Isolation Valve Outboard Exhaust Line Containment Isolation Valve Overpressure Vent Line Isolation Valve
	The CIS supplies nitrogen to the Plant Pneumatics System to support actuation of pneumatically operated valves located inside of containment.	PPS Supply Line Isolation Valve

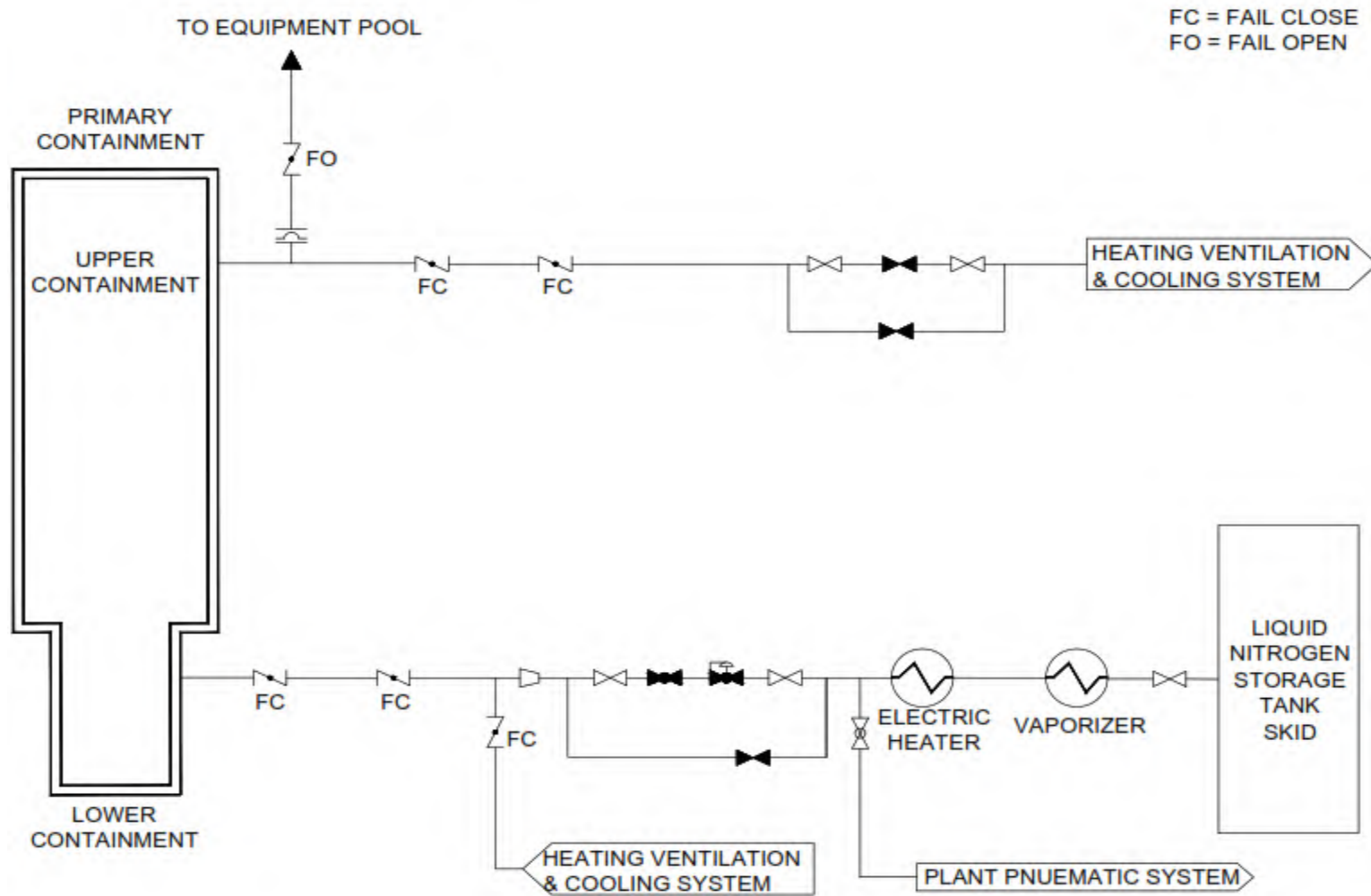


Figure 9A.4.2-1: Containment Inerting System

## **9A.5 Heating, Ventilation and Air Conditioning Systems**

The Heating, Ventilation and Cooling System (HVS) serves all areas of the power block during normal operation, with the exception of the containment which is serviced by the Containment Cooling System (Subsection 9A.5.6). HVS services containment in de-inerted modes of operation cross connected with the Containment Inerting System (Subsection 9A.4.2). The HVS maintains space design temperatures, quality of air and pressurization. It provides a controlled environment for personnel safety and comfort, and for the proper operation and integrity of equipment located in the power block.

The HVS consists of subsystems for fresh air and ventilation of each building. With the exception of the Control Building subsystem (Subsection 9A.5.2), all of the subsystems exhaust to a common plant exhaust stack during all normal operation modes. The plant exhaust stack is monitored for radiation. The Control Building is outside the radiologically controlled area and therefore the exhaust does not require radioactive monitoring.

The HVS is broken down into 6 different areas:

- Control Building
- Reactor Building
- Radwaste Building
- Turbine Building
- Plant Services Area of the Turbine Building (PLSA)
- Plant Vent Stack (PVS)

### **9A.5.1 Reactor Building Heating, Ventilation and Air Conditioning System**

#### **9A.5.1.1 System and Equipment Functions**

RB HVS, system, and equipment functions include the following:

##### **9A.5.1.1.1 Normal Functions (Non-Safety Category)**

1. Provides a controlled environment for personnel comfort and safety and equipment operation. Outside air is provided to meet the ventilation requirements for indoor air quality consistent with the requirements of ASHRAE 62.1 (Reference 9A.5.1-1).
2. Maintains designated clean areas at higher than atmospheric (positive) pressure to minimize the infiltration of outside air.
3. Maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere or to other rooms.
4. The Battery Rooms are ventilated and exhausted by the RB HVS to prevent hydrogen accumulation in the rooms.
5. Reduces the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
6. Detects and limits the introduction of airborne radioactivity into the SCR.
7. Provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

8. The HVS is designed such that failure of Non-Safety-Category function equipment does not compromise Safety Category function equipment.
9. The HVS supplies outside air to the Steel-plate Composite Containment Vessel (SCCV) via the Containment Inerting System (Subsection 9A.4.2) and exhausts the SCCV inerting gases to provide a habitable environment for maintenance personnel during outages.

#### **9A.5.1.1.2 Off-Normal Functions (Non-Safety-Category)**

RB HVS systems continue to operate during off-normal conditions until removed from service either manually or automatically.

In the event of a LOOP, the RB HVS continues to function due to its connection to the Diesel Generators which activate and provide backup power to their respective train-supported equipment. This includes the normal AHUs, their associated electric heaters, and Fan Coil Units for the Defense Line 2 (DL2) and DL4a DCIS rooms.

#### **9A.5.1.1.3 Normal Function (Safety-Category)**

The RB HVS provides required cooling to Safety Class rooms with two redundant SC3 FCUs in each room (Figure 9A.5.2-1).

#### **9A.5.1.1.4 Off-Normal Functions (Safety-Category)**

The RB HVS performs Safety-Category (TBD) functions. The RB HVS has sufficient capacity to fulfill its safety function and there are no credible single failures or operator errors that could defeat the performance of the safety function for which the system was designed.

The RB HVS includes two Safety-Category (TBD), 100% capacity pressurization fans with emergency filtration units which supply filtered outside air to the SCR which is used to pressurize the space relative to adjacent spaces, protecting the operators from hazards such as high radiation levels resulting from DBAs, release of radioactive material. These pressurization fans are provided with backup power making them available for use during a LOOP or Station Blackout (SBO).

The SCR and supporting Division 1, 2, and 3 Electrical and Switchgear rooms are provided with passive cooling. Emergency filtration and pressurization systems for the SCR are powered from the SC1 Electrical Distribution System batteries for the required 72 hours.

Tornado dampers are provided at each RB HVS penetration to protect the RB structure.

The RB HVS includes Safety-Category (TBD) supply and exhaust isolation dampers for the RB, which close automatically in the event of a fuel handling accident, isolating the RB. These dampers located at the RB boundary and the associated controls that provide the isolation signal are seismic..

The design of the HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 (Reference 9A.5.1-2) Section 7.10 as related to providing an HVAC system capable of maintaining the required environmental conditions for systems and components important to safety in all states.

Refer to Chapter 11, Subsection 11.3 for information pertaining to Gaseous Waste Management System.

The design of the RB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 as related to providing a ventilation system with a filtration system capable of the following:

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

1. Preventing unacceptable dispersion of airborne contaminants within the plant. Airborne contaminants are contained through the use of two 100% capacity RB Exhaust AHUs with HEPA filtration which take suction on the RB keeping it at negative pressure, ensuring that air flows into the RB from adjacent clean air spaces. The potential spread of airborne contamination is minimized by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.1-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases by using charcoal filters.

**9A.5.1.2 Safety Design Bases**

1. As part of Safety-Category (TBD) requirements, the RB HVS ensures that a habitable environment is maintained in the SCR for a minimum of 72 hours following an initiating event.
2. As part of Safety-Category (TBD) requirements, the RB HVS ensures that the SCR temperature does not exceed the limiting temperature during a postulated design basis accident event.
3. As part of Safety-Category (TBD) requirements, the RB HVS is designed to ensure that the SCR operators do not receive radiation exposure in excess of 0.05 Sv (5 rem) total effective dose equivalent in a single year.
4. As part of Defense Line 2, the RB HVS provides the capability to isolate ventilation to the SCR when signaled to do so from Safety-Category (TBD) Instrumentation and Control System.
5. Tornado dampers are provided for the RB HVS in support of Safety-Category (TBD) functions.
6. As part of Defense Line 2, the RB HVS provides the capability to isolate ventilation to the refuel floor when signaled to do so from Safety-Category (TBD) Instrumentation and Control System.
7. RB HVS interfaces with the Safety-Category (TBD) Instrumentation and Control System monitor and diagnostics function to monitor Safety-Category (TBD) equipment room temperature information.



8. The RB HVS battery room exhaust fans are powered from the same bus as Safety-Category (TBD) Electrical Distribution System battery chargers associated with the respective divisional room.

#### **9A.5.1.3 Description**

Figure 9A.5.1-1 depicts the Reactor Building HVAC Process Flow Diagram.

The RB, for HVAC purposes, is divided up into two main areas which includes the lower levels and the upper areas.

The RB lower levels are supplied with conditioned air provided by one of two AHUs, located on the Control Building roof. These AHUs supply air through vertical duct chases down to the lowest elevation of the RB. The RB lower-level supply AHUs also supply outside air to the Containment Inerting System (Subsection 9A.4.2) when required for containment de-inerting and to provide a habitable environment for maintenance personnel during refueling and maintenance operations.

Two 100% capacity RB exhaust AHUs with HEPA filtration take suction on the RB keeping it at negative pressure, ensuring that air flows into the RB from adjacent clean air spaces. The operating exhaust AHU discharges through ductwork to the Continuous Exhaust Air Plenum (CEAP) and to the PVS. These AHUs are provided with adjustable speed drives to be able to reduce flow comparable to reduced supply air flow to the RB. The normal lower-level exhaust ductwork also receives air and inerting gases from the CIS when required, which is filtered and exhausted to the Continuous Exhaust Air Plenum and to the PVS.

Each of the three Division Battery Rooms have two 100% capacity exhaust fans which are provided with backup emergency power, exhausting to atmosphere in the event of a Loss-of-Offsite Power through ductwork and a louver mounted in a room up above. The Battery Rooms normally exhaust to the operating RB Exhaust AHU and the PVS. The RB lower-level AHU's also supply conditioned outside air to the Containment Inerting System when needed.

The supply ductwork to each of the three Battery Rooms, (Division 1, 2 and 3 Battery Room) and the Secondary Control Room are each provided with electric duct heaters.

Supplemental cooling and air circulation is provided by fan coil units.

Heating to the RB Stairwells is supplied by electric cabinet heaters while heating to the Entry/Truck Bay is provided by an electric unit heater.

When needed, filtered outside air is supplied to the Secondary Control Room by one of two capacity Pressurization Fans and Emergency Filtration Unit. The two pressurization fans draw outside air through dedicated SC1, Seismic Category B safety-related ductwork with blast resistant openings located in the exterior walls. The two makeup AHUs draw outside air through dedicated Safety-Category 1 ductwork with blast resistant openings.

The operating SCR makeup AHU pressurization fan maintains a minimum positive pressure in the SCR with filtered makeup air, protecting the operators from potential radioactive gases. The SCR is normally provided filtered, conditioned air from the operating RB Lower-Level AHU.

The RB upper levels are supplied with conditioned air provided by AHUs located on the Plant Service Area roof. These AHUs are provided with filters, electric heating coils, and chilled water supplied cooling coils, supplying conditioned outside air to the RB upper levels. These AHUs employ adjustable speed drives to vary flow to the fuel pool as needed during refueling, and to reduce flow during reduced temperature ambient conditions.

Exhaust air from the RB lower levels is combined with exhaust air from the upper levels. Exhaust air is discharged out of the RB building by AHUs. These AHUs are provided with High Efficiency

Particulate Air (HEPA) filters, prefilters and exhaust fans. The discharge is sent to the Continuous Exhaust Air Plenum and PVS.

#### **9A.5.1.4 Component Description**

Design information related to the major components of the Reactor Building HVS is provided below.

##### **9A.5.1.4.1 Filters**

The various building filtration levels are specified within each specific Subsection. Filters meet the applicable efficiency rating as stated below. American Society of Heating, Refrigerating and Air Conditioning Engineers (ASHRAE) Standard 52.2 (Reference 9A.5.3-4) establishes a filter's Minimum Efficiency Reporting Value (MERV) and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

- Low Efficiency:           MERV 1-4 (Less than 20%)
- Medium Efficiency:      MERV 5-12 (As least 20% but less than 80%)
- High Efficiency:         MERV 13-16 (Greater than or equal to 80%)

HEPA filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.1-4) are usually rated by the dioctylphthalate) test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle-collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.1-5).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.1-6).

Each of the two SCR Emergency Makeup Filtration Units contains a charcoal filter, located between upstream and downstream banks of HEPA filters. The upstream bank protects the charcoal from clogging. The downstream bank prevents discharging charcoal fines in the SCR. The purpose of the charcoal bed is to remove potentially radioactive gases from the air stream. The charcoal bed also has properties associated with toxic gas adsorption.

##### **9A.5.1.4.2 Air Handler Units**

Each AHU consists of an inlet area, filters (as specified by the system), electric heating coils, (as required) cooling coils (as required) and the respective fans (supply or exhaust). Bag In/Bag Out AHUs and Emergency Filter Units (EFUs) are intended for use in units that contain potentially contaminated filters to facilitate filter changeout without the worker or room being exposed to the potentially contaminated filter.

##### **9A.5.1.4.3 Supply and Exhaust Fans**

The various building ventilation systems are provided with supply and exhaust fans, sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of the Air Movement and Control Association International. Fans in various areas are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

#### **9A.5.1.4.4 Heating Coils/Elements**

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, "Heating and Cooling Equipment" (Reference 9A.5.1-7).

#### **9A.5.1.4.5 Cooling Coils**

The Tubular fin type cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.2-8) and ANSI/ARI 410 (Reference 9A.5.2-9) and Underwriters Laboratory (Reference 9A.5.2-7).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to the Equipment and Floor Drain System (Subsection 9A.9.3).

#### **9A.5.1.5 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls identified in the equipment and purchase specifications.

#### **9A.5.1.6 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.1-1 for RB HVS interfaces with other equipment or systems.

#### **9A.5.1.7 System and Equipment and Operation**

##### **9A.5.1.7.1 Normal Operation**

The RB lower elevations will normally be provided heated, conditioned, and filtered once-through supply air from one of two operating AHUs located on the Control Building roof through supply ductwork. AHU cooling is supplemented by cooling provided by chilled water supplied FCUs located in rooms that require extra cooling, with FCUs being controlled by room thermostats. An electric duct heater in the supply duct to each Battery Room and to the SCR is expected to normally be operating as required to maintain SCR habitability and Battery Room temperatures as recommended by the battery manufacturer.

The RB upper elevations, consisting mainly of the fuel handling area operating deck, will normally be provided heated, conditioned, and filtered once-through supply air from one of two operating AHUs located on the Plant Service Area roof through supply ductwork. For each of the operating RB upper-level supply AHUs, the second supply AHU will normally be in standby, ready for auto-start should the operating AHU fail.

The Truck Bay, in addition to being provided conditioned air from the operating upper-level supply AHU, is heated by an electric unit heater, controlled by local thermostat.

Air will normally be exhausted from the RB upper and lower levels, including the Battery Rooms, from one of two operating exhaust AHUs discharging ultimately to atmosphere from the PVS.

For each of the RB supply and exhaust AHUs, the second AHU will normally be operational, in standby, ready for auto-start should the operating AHU fail.

##### **9A.5.1.7.2 Off-Normal Operation**

In the event of a LOOP, power is maintained to both RB exhaust AHUs, and all four RB supply AHUs and associated heating coils for the 4 supply AHUs.

In the event of a SBO, power is lost to the RB exhaust AHUs as well as the upper and lower-level supply AHUs which normally provide heating and air conditioning for the RB, as well as electric duct heaters for the Battery Rooms.

Each Battery Room, normally exhausted via the RB exhaust AHU, continues to be exhausted via one of two 100 percent capacity exhaust fans, powered from its backup power source, for the purpose of being placed into service during loss of power events. These fans are controlled by ON/OFF/AUTO-switches located in the SCR, with fan auto-starts initiated by undervoltage on the RB exhaust AHUs' power supply.

In the event of a loss of the MCR in the CB, the SCR in the RB is activated, and its dedicated HVAC system is placed into service. The emergency pressurization fans with charcoal radioactive iodine gas filtration, and backed up by battery power, will also be placed into service automatically upon a loss of power to the RB lower-level normal supply AHUs or detection of radiation, at the RB lower-level normal supply AHU intakes.

The Secondary Control Room pressurization EFUs and emergency pressurization fans, including the intake, ductwork, and power supplies are designed to operate for up to seventy-two hours without normal AC power.

In the event of a fuel handling accident on the refueling floor in the Fuel Handling Machine area, a radiation monitor alarm initiates closure of the RB isolation dampers and securing of the RB upper level supply AHUs.

#### **9A.5.1.8 Instrumentation and Control**

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control cooling water control valves, electric heating elements, or control fan speeds.
2. Moisture element in the SCR room to engage RB lower-level supply AHU dehumidification control.
3. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
4. Differential pressure transmitters to monitor pressure drop across AHU filters.
5. Intake, exhaust, and return air damper position monitoring
6. Duct mounted smoke detectors to shut down fans as required

Manual initiation and shutdown of the RB HVS are provided at the Temperature Control Panel.

The following control features are implemented in the RB HVS:

1. The RB AHUs, FCUs, and exhaust fans are operated from the RB temperature control panels. An AHU start request automatically opens all associated shutoff dampers. A damper operating failure (any associated damper not 100% open while fan is running) automatically trips that particular AHU.
2. Any AHU trip automatically closes all its associated shutoff dampers.
3. Any AHU started in standby because of a low air flow condition automatically trips the AHU that generated the low flow condition after proof that the standby AHU is running. To avoid spurious standby starts and AHU trips, all air flow logic incorporates time delays to allow fans to generate steady-state flow.

4. Duct mounted temperature elements monitor the operation of the RB AHUs.
5. Local wall mounted temperature elements in the SCR and Battery Rooms control the electric duct heaters in the supply air ducts to those rooms.
6. Control valves in RB AHU cooling coil supply return waterlines are modulated to maintain a constant coil leaving air temperature. If the outside air temperature is less than a predetermined temperature, the control valve supplying the RB supply AHU cooling coils is 100% closed; otherwise, the valve is modulated.
7. The FPS provides a signal to shut down the RB AHUs in the event of a fire. The Battery Room exhaust fans continue to operate, preventing a potentially explosive hydrogen concentration from occurring. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.
8. The Battery Room exhaust fans operate off timers. Failure of any Battery Room exhaust fan is annunciated via a common HVS alarm in the MCR.
9. The lower RB supply AHU intakes are monitored for toxic gas and radiation. If toxic gas or radiation are detected, the lower RB supply AHU turns off and the isolation dampers shut. Additionally, the SCR normal supply and exhaust isolation dampers shut, the SCR pressurization fans energize, supplying the SCR with filtered outside air.
10. Upon high radiation detection in the Operating Deck area of the RB, the RB upper level supply and exhaust isolation dampers automatically close, and associated supply AHUs are secured, isolating supply and exhaust air to/from that area. These dampers along with associated actuators are SC3, seismically qualified, and fail shut on a loss of power or signal.

Instrumentation is designed to support human performance. Alarms and notifications are clearly labeled and identifiable.

The RB HVS system is installed and located in physically accessible locations whenever possible. Otherwise, the means by which operators and personnel can access equipment is provided.

The MCR is provided with a Master Alarm indicator from each temperature control panel. The Master Alarm indicator prompts operators to conduct further review at the local temperature control panel. The local temperature control panel provides additional information and the means of resetting alarm conditions on a return-to-normal transition.

Any Safety-Category functions or other required functions happen automatically with notification provided to operators.

#### **9A.5.1.9 Monitoring, Inspection, Testing, and Maintenance**

##### **9A.5.1.9.1 System Level Requirements**

A positive pressure test is performed on RB HVS serving contaminated areas in the RB. This test is performed as required.

Plant-level HVAC inspection and maintenance is conducted in accordance with ASME N511 (Reference 9A.5.1-10).

##### **9A.5.1.9.2 Component Level Requirements**

Air handlers are field tested as required per ASME N511, (Reference 9A.5.1-10).

Charcoal beds used for filtration of radiological effluents support meeting ALARA requirements as discussed in Chapter 12, Section 12.1.

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

HEPA filters conform to the requirements of ASME of AG-1 (Reference 9A.5.1-4).

#### **9A.5.1.10 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.1.11 Performance and Safety Evaluation**

The RB HVS has sufficient capacity to fulfill its safety function and there are no credible single failures or operator errors that could defeat the performance of the safety function for which the system is designed. The RB HVS is designed with redundancy to assure that subsystems are normally available during all modes of plant operation, including startup and shutdown.

The RB HVS ensures that a habitable environment is maintained in the SCR for a minimum of 72 hours following an initiating event. The SCR and supporting Division 1, 2, and 3 Electrical and Switchgear rooms are provided with passive cooling systems powered from the Safety-Category (TBD) Electrical Distribution System batteries for the required 72 hours. The supplemental passive cooling system also ensures that the SCR temperature does not exceed limiting temperature limits during a postulated design basis accident.

The RB HVS is designed to preclude SCR operators from receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent in a single year by utilizing two Safety-Category (TBD), 100% capacity pressurization fans which supply filtered outside air to the SCR which pressurizes the space relative to adjacent spaces. This design feature protects operators from radiation associated with a Design Basis Accident.

The RB HVS includes supply and exhaust isolation dampers which close automatically in the event of a fuel handling accident, isolating the RB upper levels. These dampers are located at the RB boundary. In addition, the RB HVS provides the capability to isolate ventilation to the SCR. In both cases, the associated controls that provide the isolation signal is supplied from the Safety-Category (TBD) Instrumentation and Control System.

Tornado dampers are provided for the RB HVS supporting Safety-Category (TBD) functions.

The Safety-Category (TBD) Instrumentation and Control System monitors and provides diagnostic functions relative to RB HVS Safety-Category (TBD) equipment room temperature.

Battery room exhaust fans are powered from the same bus as the Safety Class 1 Electrical Distribution System battery chargers associated with the respective divisional room. This requirement prevents hydrogen production from battery charging operations if the exhaust fans stop working due to a loss in the bus power.

#### **9A.5.1.12 References**

9A.5.1-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.5.1-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.1-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.1-4 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.1-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.1-6 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.1-7 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.1-8 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.1-9 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.1-10 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.5.1-1: Reactor Building Heating, Ventilation and Air Conditioning System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks	At the HVS equipment
Process Radiation and Environmental Monitoring System	The PREMS provides continuous radiation monitoring of the HVS	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water for cooling of the HVAC FCUs and AHUs	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SCR emergency fans and electric duct heaters	At the HVS equipment and heaters
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads	At the HVS equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning Unit cooling coils.	At the air handler or fan coil unit flange
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides Safety-Category (TBD) control to the SCR EFUs and associated dampers.	At the RB HVS equipment
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides Safety-Category (TBD) control to RB supply and exhaust isolation dampers, TB supply and exhaust dampers, and CRE EFUs.	At the RB HVS equipment
Safety-Category (TBD) Electrical Distribution System	Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs	At the RB HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to RB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Containment Inerting System	Containment Inerting System discharges containment inerting gases to the Continuous Exhaust Air Plenum /Plant Vent Stack during purging. RB HVS provides outside air supply to CIS for containment de-inerting	At the Containment Inerting System/RB HVS piping interface
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring	At the RB HVS equipment

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

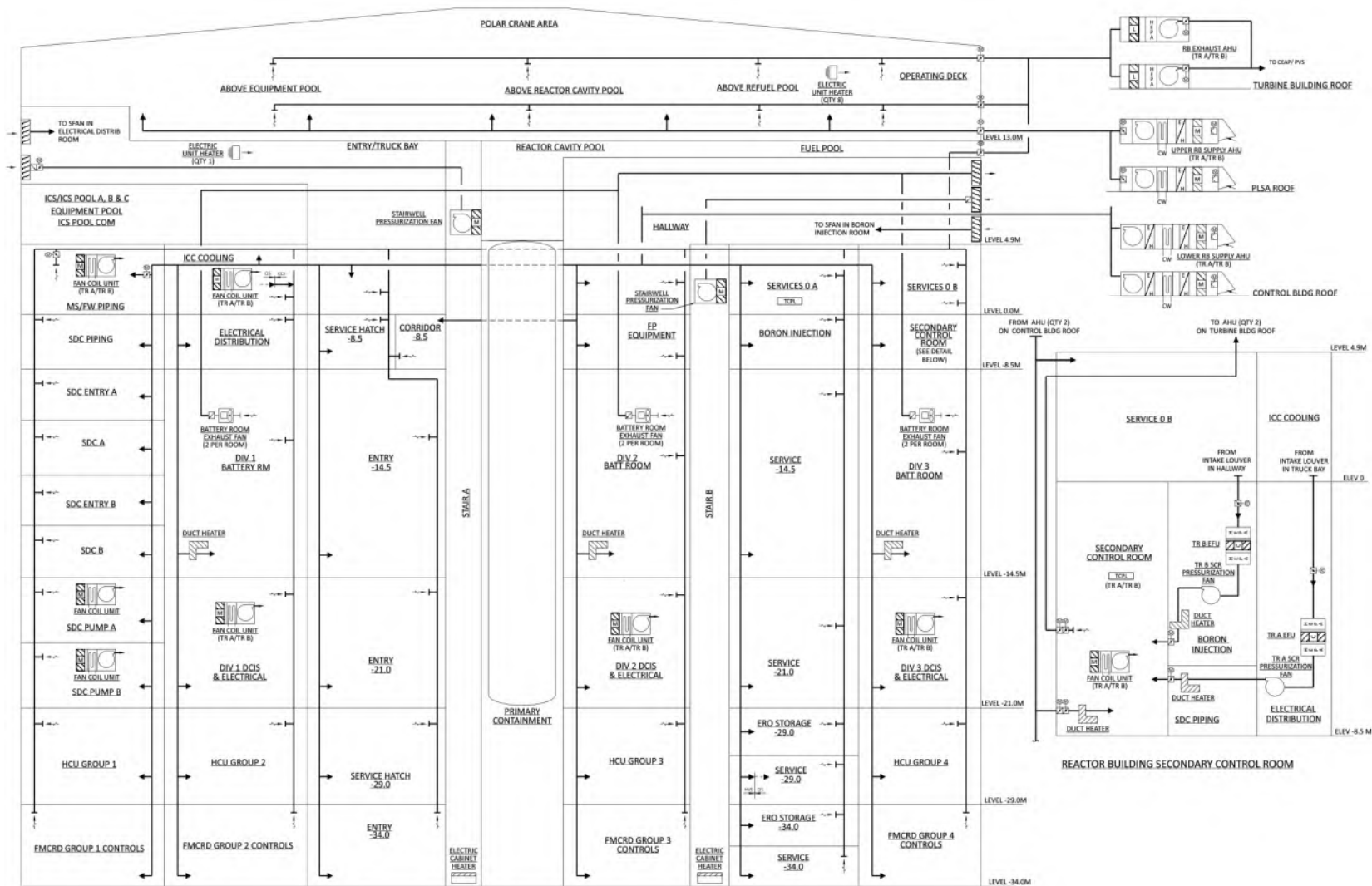


Figure 9A.5.1-1: Reactor Building HVAC

## **9A.5.2 Control Building Heating, Ventilation and Air Conditioning System**

### **9A.5.2.1 System and Equipment Functions**

#### **9A.5.2.1.1 Normal Functions (Non-Safety Category)**

1. Provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.2-1).
2. Provides a controlled environment for operation of equipment in the Power Block during normal, startup and shutdown operations.
3. Provides isolation features to support testing and maintenance.
4. The CB HVS supply ductwork to all the Battery Rooms, are provided electric duct heaters for more precise temperature control in those rooms.
5. The Battery Rooms are ventilated and exhausted to maintain hydrogen levels below required limits in the rooms.
6. The CRE in the Control Building is maintained at a higher pressure than surrounding areas with the CRE EFUs in service.
7. Detects and limits the introduction of airborne hazardous materials (toxic gas, radioactivity, or smoke) into the Control Room in the Control Building.
8. The HVS is designed such that failure of Non-Safety-Category function equipment does not compromise Safety-Category function equipment.
9. Provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the power block to the outside atmosphere in the event of a fire.

#### **9A.5.2.1.2 Off-Normal Functions (Non-Safety Category)**

The Control Building HVS performs several Non-Safety-Category functions during off-normal conditions. The CB HVS continues to operate during off-normal conditions until removed from service either manually or automatically.

CB HVS equipment that is normally in standby that is required to function during off-normal events include the following:

1. In the event of a fire outside the CB, the outside air through the normal operating AHU is secured and the operating AHU continues to operate in recirculation mode only.
2. In the event of a fire inside the Control Building but outside of the Control Room Envelope (CRE), a Control Room Envelope Emergency Filtration Unit can be placed in-service, thereby supplying the CRE in pressurization mode with the CRE isolated.
3. The CB is provided ventilation air during toxic gas events as follows. The CB toxic gas filtration units operate automatically when toxic gas is detected at the CB AHU outside air intakes. Normal outside air supply to the operating CB AHU isolates and the toxic gas filtration units discharge damper opens, allowing the associated toxic gas filtration unit to supply filtered outside pressurized air to the entire Control Building through the normal CB supply AHU which continues to operate during a toxic gas event. The toxic gas filtration units are equipped with HEPA and charcoal filters to remove toxic and radioactive gases and material from the outside air supply to the normal CB AHUs.

4. In the event of high radiation detected at the CB normal air intake, the CB normal supply AHU isolates and a CRE-EFU unit auto starts to provide pressurization air to the CRE. CRE-EFU contain charcoal filtration permitting removal of radioactive iodine.
5. The CB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Control Building.
6. Battery room exhaust fans continue to operate regardless of the operational mode of the CB AHUs, or CRE EFUs. Battery room exhaust fans operate on a timer to only energize after a set period of time (determined by the offgassing rate of the batteries) and only to be energized for the minimum time needed to reduce H<sub>2</sub> concentration to below 1%.
7. In the event of a buildup of smoke inside the CB requiring a purge, two exhaust fans are manually operated to remove smoke. During this time, the AHUs providing supply operate as 100% outside air with no recirculation until such time as the smoke has been sufficiently removed from the building.
8. In the event of a LOOP, the Diesel Generators activate and provide backup power to their respective train-supported equipment. This includes the normal AHUs, Control Room Envelope Emergency Filter Units including their associated electric heaters, and FCUs for the DL2 and DL4A rooms.

**9A.5.2.1.3 Normal Functions (Safety Category)**

1. The CB HVS provides required cooling to the DL4a Room in the CB with two redundant Safety-Category (TBD) FCUs.
2. The CB HVS provides required cooling to the Safety-Category (TBD) Instrumentation and Control System DL2 Room A and Safety-Category (TBD) Instrumentation and Control System DL2 Room B with redundant Safety-Category (TBD) FCUs in each room.

**9A.5.2.1.4 Off-Normal Functions (Safety Category)**

SC3 Functions:

1. The normal CB outside air intakes are monitored for airborne radioactivity. Upon detection of a high radiation condition, the operating normal supply AHU will de-energize, auto-start of the standby unit is defeated, and the CRE goes into isolation mode with normal CB supply and return isolation dampers closing. In addition CRE EFUs automatically start, pressurizing the CRE with filtered supply air, utilizing a mixture of approximately 90% return air from the CRE. The 10% outside air passes through the CRE-EFU with potential radioactive iodine being removed via charcoal filtration.
2. Tornado dampers are provided at each CB HVS outside air interface to protect the CB structure.

The design of the CB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2, Section 8.11.2 (Reference 9A.5.2-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of airborne contaminants within the plant by monitoring the normal CB outside air intakes for radioactivity. Upon detection of radioactivity the CRE goes into isolation mode with normal CB supply and return isolation dampers going closed. CRE-EFU automatically starts, pressurizing the CRE

with filtered supply air, utilizing a mixture of approximately 90% return air from the CRE. The 10% outside air passes through the CRE-EFU with potential radioactive iodine being removed via charcoal filtration.

2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases by using charcoal filters that have properties for toxic gas adsorption.

#### **9A.5.2.2 Safety Design Bases**

The safety design bases associated with the Control Building HVS include the following:

1. The Safety-Category (TBD) CB HVS ensures that a habitable environment is maintained in the MCR.
2. The CB HVS maintains a slight positive pressure in the CRE during emergency events.

#### **9A.5.2.3 Description**

Figure 9A.5.2-1 depicts the Control Building HVAC Process Flow Diagram.

The Control Building HVS subsystems are comprised of redundant trains of equipment that support all CB HVS functions during normal operating modes.

The Battery Room exhaust fans in the Control Building are backed up by batter power.

The CRE in the Control Building is maintained at a higher pressure than surrounding areas with the CRE-EFU in service.

The Control Room HVS detects and limits the introduction of airborne hazardous materials (radioactivity, or smoke) into the Control Room in the Control Building. This system is designed with the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Smoke control and removal functions are in accordance with National Fire Protection Association (NFPA); NFPA 92A and CSA N293/293S1 and National Building Code of Canada (NBCC) and National Fire Code of Canada as described in Subsection 9A.6).

The MCR is provided filtered, conditioned air from two 100% capacity AHUs via supply and return ductwork. Refer to Chapter 6, Subsection 6.4 for information pertaining to Control Room Habitability.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

The battery rooms are each provided with two 50% capacity exhaust fans which operate based off timers. Other exhaust fans are provided for the restroom/shower areas and the control room restroom/janitor's closet.

The CB is a single-story building. Some rooms contain significant quantities of electrical equipment and associated heat dissipation loads. Other rooms, such as the MCR, Radiation Protection offices, and Conference Room, contain areas of less heat density. The Battery Rooms, Restrooms, Break Room, and Janitor's Closet are also low heat density rooms and will be served by exhaust fans exhausting to the outdoors.

The CB is normally air conditioned and heated by 2 x 100% capacity chilled water supplied AHUs located on the CB roof, discharging through supply ductwork and a return plenum. Approximately 10% fresh makeup air is brought into the operating AHU. In addition to the chilled water coils, the AHUs contain electric heating coils, humidification/dehumidification features, filtration, and isolation dampers. Supply ductwork upstream of the MCR and offices is provided with Volume Control Units to vary supply air flow based on thermostatic control. AHU cooling in the high heat load rooms is supplemented by chilled water supplied FCUs with a Train A and a Train B FCU located in each of these rooms.

The Battery Rooms are maintained at the battery manufacturer recommended room temperature. Battery Room temperatures are maintained via Battery Room supply duct electric duct heaters and thermostats located in the rooms.

The outside air intake is instrumented to analyze for toxic gases and for airborne radioactivity as applicable. To accommodate MCR habitability contingencies the following additional AHUs are provided for the CB:

1. CRE-EFU that operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE goes into isolation mode with normal CB supply and return air dampers going closed. A mixture of approximately 90% return air from the CRE and 10% outside air will pass through the EFU with potential radioactive iodine being removed via charcoal filtration. EFU operation maintains the CRE slightly pressurized. The operating CB normal supply AHU de-energizes upon high radiation detection at the intake and auto-start of the standby unit is defeated. Battery Room exhaust fans continue to operate based off timers.
2. CB toxic gas filtration units that operate automatically when toxic gas has been detected at the CB AHU outside air intakes. In this event, normal outside air supply to the operating CB AHU isolates and the toxic gas filtration unit discharge damper opens, allowing the associated toxic gas filtration unit to supply pressurization air to the CB through the normal CB supply AHU, which continues to operate during a toxic gas event.

The CB normal supply AHUs are provided ASD to be able to reduce flow as conditions change.

CRE isolation and CRE-EFU operation provide a pressurized envelope with respect to adjacent spaces, maintaining CRE habitability during an airborne radiation event. Toxic gas filtration unit operation maintains the entire CB pressurized, maintaining CB habitability during a toxic gas event.

Two smoke exhaust fans are provided for the CB. During recovery from a fire, smoke is exhausted from the CB by operating a CB normal supply AHU in 100% outside air mode in conjunction with operation of two CB smoke exhaust fans.

The CB HVS is provided tornado rated dampers as necessary to withstand high wind events.

#### **9A.5.2.3.1 Component Description**

##### **Filters**

The various building filtration levels are specified within each specific Subsection. Filters meet the applicable efficiency rating as stated below. American Society of Heating, Refrigerating and Air Conditioning Engineers Standard 52.2 (Reference 9A.5.2-4) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

- Low Efficiency:           MERV (Less than 20%)
- Medium Efficiency:   MERV (As least 20% but less than 80%)
- High Efficiency:       MERV (Greater than or equal to 80%)

HEPA Filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.2-4) are usually rated by the dioctylphthalate test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle-collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.2-5).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.2-6).

##### **Air Handler Units**

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1-2019 Code on Nuclear Air and Gas Treatment (Reference 9A.5.2-7).

##### **Supply and Exhaust Fans**

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of AMCA. Fans in various areas are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

##### **Heating Coils/Elements**

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (Reference 9A.5.2-8).

##### **Cooling Coils**

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33 (Reference 9A.5.2-9), Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils, and ANSI/ARI 410 (Reference 9A.5.2-10) and Underwriters Laboratory (Reference 9A.5.2-8).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Chapter 9, Subsection 9A.9.3).

#### **9A.5.2.4 Materials**

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

#### **9A.5.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.2-1 for Control Building HVS interfaces with other equipment or systems.

#### **9A.5.2.6 System and Equipment Operation**

##### **9A.5.2.6.1 CB HVS Normal Operations**

The CB HVS is provided heated, conditioned, and filtered supply air from one of two operating AHUs through supply ductwork, with approximately 90% of the air recirculated back to the units. The second AHU is in standby, ready for auto-start should the operating unit fail. The remaining 10% of the supply air is made up by outside air brought into the operating AHU located on the CB roof. The two exhaust fans provided for each Battery Room are operating based off timers, exhausting to the outdoors. Exhaust fans for the restrooms, break room, and janitors' closet are all expected to be operating continuously, exhausting to the outdoors. An electric duct heater in the supply duct to each Battery Room is normally operating as required to maintain Battery Room temperatures near the optimal battery temperature as recommended by the battery manufacturer.

##### **9A.5.2.6.2 CB HVS Off-Normal Operations**

In the event of an SBO, power is lost to the CB AHUs which normally provide heating and air conditioning for the CB, as well as electric duct heaters and cabinet heaters. The restroom, break room, and janitors closet exhaust fans become inoperable. The Control Room operators relocate to the SCR in the RB if the MCR becomes uninhabitable.

In the event of a CB normal outside air intake radiation monitor alarm, the operating normal supply AHU de-energizes and any associated toxic gas filtration units also de-energize, and the same train CRE-EFU automatically starts. The CRE isolation dampers close in conjunction with the start of the CRE-EFU. Diesel generator backup power is provided to the CRE-EFU supply fans and associated heating coils. Restroom, break room, and janitors closet exhaust fans become inoperable. The Battery Room exhaust fans continue to operate off timers using backup power, preventing possible accumulations of off-gassed hydrogen to the rooms; the timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. Electric duct and cabinet heaters are de-energized.

In the event of a CB normal outside air intake toxic gas detection, the operating normal AHU normal outside air intake closes, and the same train toxic gas filtration unit automatically starts and its associated motor control dampers automatically open, providing an alternate source of filtered outside air for CB pressurization. Diesel Generator backup power is provided to the CB normal supply AHU fans and associated heating coils, as well as the associated Toxic Gas Filtration Unit fan. Restroom, break room, and janitors closet exhaust fans become inoperable. The Battery Room exhaust fans continue to operate off timers using backup power, preventing possible accumulations of off-gassed hydrogen to the rooms; the timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. Electric duct and cabinet heaters are de-energized. Positive pressure is maintained in the CB by way of providing outside air through the normal supply AHU, the toxic gas filtration unit, or the CRE-EFU.

#### **9A.5.2.7 Instrumentation and Control**

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control cooling water control valves, electric heating elements, or control fan speeds."



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

2. Moisture element in the MCR to engage the CB Normal Supply AHU humidification and dehumidification control.
3. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
4. Differential pressure transmitters to monitor pressure drop across AHU filters.
5. Intake, exhaust, and return air damper position monitoring.
6. Duct mounted smoke detectors to shut down fans as required.

The HVS subsystems have dedicated control panels that regulate the temperature of the spaces to within specified ranges.

Manual initiation and shutdown of the CB HVS is provided from the CB Temperature Control Panel located in the Technical Support Center.

The following control features are implemented in the CB HVS:

1. The CB AHUs, supply fans, and exhaust fans have the capability of being remotely started from the Emergency Operations Center temperature control panel. Bathroom exhaust fans are wired directly from lighting panel, locally controlled with no DCIS. To avoid air infiltration from the surrounding areas, all rooms in the CB with the exception of the janitor's closet, restrooms, breakroom, Battery Rooms and stairwell are positively pressurized by the introduction of filtered outside air and by maintaining the pressure boundary integrity.
2. Battery Room exhaust fans operate off timers. Failure of any Battery Room exhaust fan is annunciated via a common CB HVS alarm in the MCR.
3. Any fan started in standby because of a low airflow condition automatically trips the fan that generated the low flow condition after confirmation that the standby fan is running. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
4. The CB differential pressure transmitters are used to monitor the pressure in the MCR and alarm low differential pressure.
5. Control valves located at the cooling coils modulate the flow of chilled water in AHUs in response to air temperature signals from the CB HVS subsystems .
6. Local temperature switches cycle ON/OFF electric cabinet heaters cycle "ON/OFF."
7. The humidifiers in the CB AHU are controlled by the moisture elements located in the MCR.
8. The outside air normal intake dampers to the CB AHUs cannot be opened if smoke, toxic gas, or radiation is detected.
9. The CB recirculation air damper cannot be closed if radiation or smoke is detected in the outside air intake.
10. Upon detection of toxic gas at an outside air normal intake, the normal outside air intake damper closes and the associated toxic gas filtration unit and its associated suction and discharge dampers open, providing an alternate, filtered source of outside makeup air to the operating normal supply AHUs.

11. Upon receipt of an outside air normal intake radiation monitor alarm, the operating normal supply automatically stop, and the associated same train automatically starts, pressurizing the CRE and protecting CRE staff from possible radioactive iodine.
12. The Fire Protection System provides a signal to shut down the CB normal and most HVAC fans in the event of a fire. The Battery Room exhaust fans continue to operate, preventing a potentially explosive hydrogen concentration from occurring. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.

#### **9A.5.2.8 Monitoring, Inspection, Testing, and Maintenance**

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.2-11).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

##### **Component Level Requirements**

Charcoal beds used for filtration of radiological effluents meet ALARA requirements as discussed in Chapter 12, Section 12.3.

#### **9A.5.2.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.2.10 Performance and Safety Evaluation**

In the event of a LOOP, diesel generator backup power is provided to the CB AHU supply fans. In the event of an SBO, power is lost to the CB AHUs. However, for both the LOOP and SBO cases the Battery Room exhaust fans continue to operate off timers using backup battery power in order to prevent potential accumulations of off-gassed hydrogen to the Battery Rooms. The timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. In the event the MCR becomes uninhabitable the Control Room operators relocate to the SCR in the RB.

CB HVS design includes:

1. Redundant Safety-Category (TBD) FCUs which provides cooling to the DL4a room.
2. Redundant Safety-Category (TBD) FCUs which provides required cooling to the Safety-Category (TBD) Instrumentation and Control System DL2 Room A and Safety-Category (TBD) Instrumentation and Control System DL2 Room B.

The RB HVS includes Safety-Category (TBD), 100% capacity pressurization fans with emergency filtration units that supply filtered outside air to the SCR needed to pressurize the space relative to adjacent spaces, protecting the operators from hazards such as release of radioactive materials, fire, or smoke in the RB. The pressurization fans are provided with backup power making them available for use during a LOOP or SBO.

#### **9A.5.2.11 References**

9A.5.2-1 ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.5.2-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.2-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.2-4 ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.2-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.2-6 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.2-7 ASME AG-1, "Code on Nuclear Air and Gas Treatment."
- 9A.5.2-8 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.2-9 ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.2-10 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.2-11 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

**Table 9A.5.2-1: Control Building Heating, Ventilation and Air Conditioning System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation and toxic gas monitoring of the HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs and AHUs.	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SCR emergency pressurization fans and electric duct heaters.	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs.	At the HVS equipment
Non-Safety Electrical Distribution System	The Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads.	At the HVS equipment
Water Gas and Chemical Pads	Provides clean water supply used for space humidification if needed. CB AHU & FCU condensate will be collected in drains.	At the HVS equipment
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the HVS equipment

**SIMPLIFIED DIAGRAM: CONTROL BUILDING HVAC**

**WOMEN'S ROOM EXHAUST FAN** **MEN'S ROOM & C EXHAUST FAN**

9A-150

### **9A.5.3 Radwaste Building Heating, Ventilation and Air Conditioning Systems**

The Radwaste Building contains room-temperature tanks and small to medium-sized pumps. The RWB is not normally occupied. However, plant personnel access and spend time in the Chem Lab and dress out areas within the RWB. As such, most of the building is ventilated to five air changes/hour.

The RWB Chem Lab is provided an electrical cabinet heater and a Fan Coil Unit to supplement AHU HVAC needs. The Chem Lab fume hood exhaust to the RWB exhaust AHU suction, with final discharge to the PVS.

#### **9A.5.3.1 System and Equipment Functions**

##### **9A.5.3.1.1 Normal Functions (Non-Safety Category)**

1. The RWB HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.3-1).
2. The RWB HVS provides a controlled environment for the proper operation and integrity of equipment in the RWB during normal, startup and shutdown operations.
3. The RWB HVS provides necessary isolation features to support testing and maintenance.
4. The RWB HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
5. The RWB HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
6. The RWB HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure including the Charcoal Adsorber Vault.
7. The RWB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

##### **9A.5.3.1.2 Off-Normal Functions (Safety-Category)**

The RWB HVS does not perform any Safety-Category functions.

The Radwaste Building HVS is not required to operate during or after a design basis event. The design of the RWB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.3-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of airborne contaminants within the plant through the use of the exhaust AHUs to maintain a negative pressure in clean areas, resulting in air transferring from clean to "dirty" areas."
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in

Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.

3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the RWB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

#### **9A.5.3.1.3 Normal Function Safety-Category**

The RWB HVS does not perform any Safety-Category functions.

#### **9A.5.3.1.4 Off-Normal Function Non-Safety-Category**

The Radwaste Building HVS no longer receives chilled water support during off-normal conditions and is removed from service either manually or automatically.

### **9A.5.3.2 Safety Design Bases**

The Radwaste Building HVS does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

### **9A.5.3.3 Description**

Figure 9A.5.3-1 depicts the Radwaste Building HVAC Process Flow Diagram.

RWB HVS is comprised of two 100% capacity supply AHUs and two 100% capacity exhaust AHUs with HEPA filtration, functioning in a push-pull manner. In general, clean filtered outside air, heated when needed, is supplied through ductwork to the lobby and Chem Lab, pressurizing those areas. The exhaust AHUs take suction on the potentially contaminated tank and pump rooms, creating negative pressures in those spaces, resulting in air transferring from clean to "dirty" areas." ASDs are provided for each of the four AHUs, permitting air flow to be ramped down in the wintertime, saving on heating load. Cooling in the lobby area and the Chem Lab is supplemented by chilled water supplied FCUs. The Chem Lab is also provided a small cabinet heater, and the lobby area is provided with electric unit heaters for exterior wall heating.

During recovery from a fire, smoke is exhausted from the RWB by operating the normal supply AHUs in 100% outside air mode in conjunction with the RWB exhaust AHUs.

#### **9A.5.3.3.1 Component Description**

##### **Filters**

RWB HVS filters meet the applicable efficiency rating as stated below. ASHRAE Standard 52.2 (Reference 9A.5.3-5) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Low Efficiency:	MERV 1-4 (Less than 20%)
Medium Efficiency:	MERV 5-12 (As least 20% but less than 80%)
High Efficiency:	MERV 13-16 (Greater than or equal to 80%)

HEPA filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.3-5) are usually rated by the dioctylphthalate test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle- collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.3-6).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.3-7).

### **Air Handler Units**

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.3-8).

### **Supply and Exhaust Fans**

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the applicable standards of AMCA. Fans are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

### **Heating Coils/Elements**

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (9A.5.3-6).

### **Cooling Coils**

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.3-9), and ANSI/ARI 410 (Reference 9A.5.3-10) and Underwriters Laboratory, (Reference 9A.5.3-11).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Subsection 9A.9.3) subsystem.

#### **9A.5.3.4 Materials**

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

#### **9A.5.3.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.3-1 for Radwaste Building HVS interfaces with other equipment or systems.



### **9A.5.3.6 System and Equipment Operation**

#### **9A.5.3.6.1 Normal Operations**

The RWB will normally be provided heated, filtered once-through supply air from two 50% capacity operating AHUs located on the second elevation of the RWB through supply ductwork. Outside air will be drawn in the RWB supply AHUs through suction ductwork and intake louvers located in the nearby exterior wall.

Air is exhausted from the RWB through two 50% capacity operating exhaust AHUs located on the RWB roof, which discharge to atmosphere from the PVS. AHU cooling provided by ventilation air is supplemented by cooling provided by FCUs. FCUs are controlled by room thermostats. AHU heating is supplemented by electric unit heater.

The RWB Chem Lab is provided an electrical cabinet heater and an FCU to supplement AHU HVAC needs, both are controlled by local thermostats. The Chem Lab fume hood exhausts to RWB exhaust AHU suction, with final discharge to the PVS.

#### **9A.5.3.6.2 Off-Normal Operations**

In the event of a LOOP or SBO, power is lost to all the RWB HVS equipment including the supply and exhaust AHUs, FCUs, electric unit heaters, and the Chem Lab cabinet heater and fume hood.

#### **9A.5.3.7 Instrumentation and Control**

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control electric heating elements, or control fan speeds.
2. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
3. Differential pressure transmitters to monitor pressure drop across AHU filters.
4. Intake, exhaust, and return air damper position monitoring.

Duct mounted smoke detectors to shut down fans as required.

Manual remote initiation and shutdown of the RWB HVS is provided from the RWB temperature control panel located in the Chem Lab.

The following control features are implemented in Radwaste Building HVS:

Controls for RWB HVS supply and exhaust AHUs are located in the RWB Chem Lab. All RWB AHUs have the capability of being manually started or stopped from the RWB temperature control panel.

RWB HVS AHU Operation and Interlocks:

1. Failure to start an AHU automatically starts the other AHU
2. An AHU cannot start until its shutoff dampers are confirmed 100% open
3. An AHU start request automatically opens as associated shutoff damper
4. Low flow at the inlet of the AHU trips the AHU. AHU trip initiates startup of the standby AHU
5. Redundant AHU has a "standby" or "auto" mode of operation
6. An AHU trip automatically closes its associated dampers

7. The Fire Protection System (Section 9A.6) provides a signal to shut down the RWB AHUs and most HVAC fans in the event of a fire. Smoke removal also is also performed in conjunction with the Fire Protection System. Duct mounted smoke detectors provide a smoke detected signal to the Fire Protection System local fire protection panels.

Radwaste Building HVS unit heater operation and interlocks:

- A unit heater is started automatically by a local thermostat when heating is needed.

#### **9A.5.3.8 Monitoring, Inspection, Testing, and Maintenance**

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.3-12).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

#### **9A.5.3.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.3.10 Performance and Safety Evaluation**

The radwaste building HVAC system does not perform any Safety-Category functions and therefore requires no nuclear safety evaluation.

Operational failure of any single unit of the RWB HVS does not prevent Safety Class equipment from performing any Safety-Category function.

#### **9A.5.3.11 References**

- 9A.5.3-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."
- 9A.5.3-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.3-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.3-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."
- 9A.5.3-5 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.3-6 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.3-7 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.3-8 ASME AG-1, "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.5.3-9 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.3-10 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.3-11 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.3-12 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

**Table 9A.5.3-1: Radwaste Building Heating, Ventilation and Air Conditioning System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation monitoring of the RWB HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs and AHUs.	At the HVS equipment
Non-Safety Electrical Distribution System	The Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the RWB HVS Non-Safety Category electrical loads.	At the HVS equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning cooling coils.	At the air handler or fan coil unit flange
Fire Protection System	Fire Protection System provides start/stop signals to RWB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the RWB HVS equipment

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

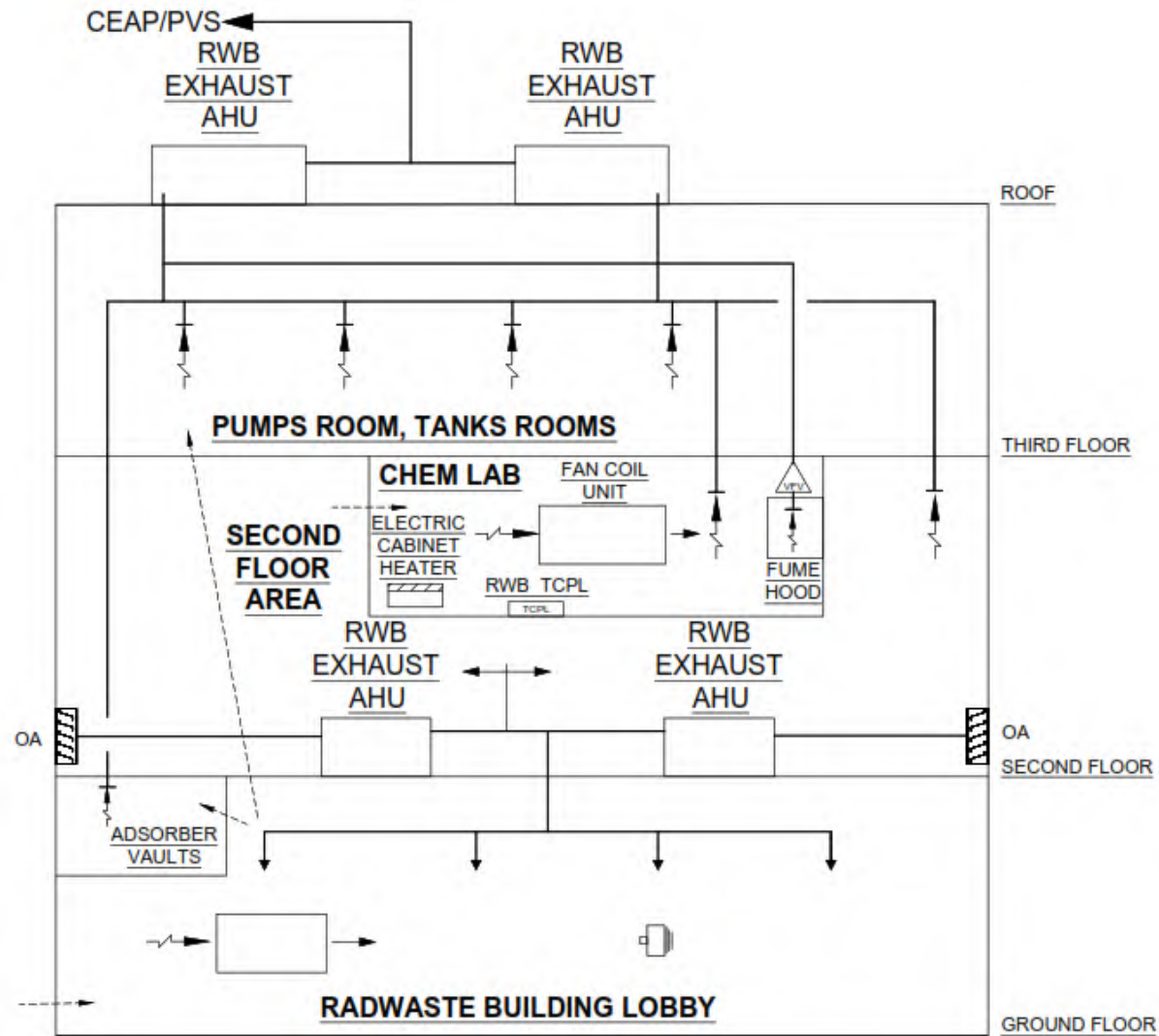


Figure 9A.5.3-1: Radwaste Building HVAC

## **9A.5.4 Turbine Building, Heating, Ventilation and Air-Condition System**

### **System and Equipment Functions**

#### **9A.5.4.1 System and Equipment Functions**

##### **9A.5.4.1.1 Normal Functions (Non-Safety Category)**

1. The TB HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.4-1).
2. The TB HVS provides a controlled environment for the proper operation and integrity of equipment in the TB during normal, startup and shutdown operations.
3. The TB HVS provides redundant active components to increase reliability, availability, and maintainability of the ventilation system.
4. The TB HVS provides isolation features to support testing and maintenance.
5. The TB HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
6. The TB HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
7. The TB HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure
8. The TB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.
9. The TB Battery Room is ventilated and exhausted to maintain hydrogen levels below required limits in the room.
10. The TB Continuous Exhaust Air Plenum collects and mixes potentially radioactive discharge air from various buildings and releases the mixed air to atmosphere via the PVS.

##### **9A.5.4.1.2 Normal Functions (Safety-Category)**

The TB HVS provides a controlled environment for the proper operation and integrity of Safety Class instrumentation located in the TB during normal, startup and shutdown operations.

##### **9A.5.4.1.3 Off-Normal Functions (Non-Safety-Category)**

TB HVS systems continue to operate during off-normal conditions until removed from service either manually or automatically.

##### **9A.5.4.1.4 Off-Normal Functions (Safety-Category)**

1. The TB HVS includes supply and exhaust isolation dampers for the TB, which close automatically in the event of a TB high radiation input from the PREMS (Subsection 9A.3.1), isolating the TB.
2. Tornado dampers are provided at the TB Diesel Generator room for the outside air intake and the DG heat exhaust louver to protect the DG structure.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

The TB HVS is not required to operate during or after a design basis event. The design of the TB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.4-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of all airborne contaminants within the plant using air flow to effect transfer of air from clean areas to potentially contaminated areas.
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the TB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

#### **9A.5.4.2 Safety Design Bases**

The TB HVS is not required to operate during or following a Design Basis Accident. The safety design bases associated with the TB HVS is provided below:

1. Safety-Category (TBD) Tornado dampers are provided at the TB DG room for the Outside Air intake and the DG heat exhaust louver to protect the DG structure.
2. As part of Defense Line 2, the TB HVS provide the capability to isolate ventilation to the TB when signaled to do so from the Safety-Category (TBD) Instrumentation and Control System.
3. TB HVS maintains temperatures in the TB required to support operation of Safety Class instrumentation.

#### **9A.5.4.3 Description**

Figure 9A.5.4-1 depicts the Turbine Building HVAC Process Flow Diagram.

The TB is a large building with significant cooling loads. It is divided into two main areas: The potentially contaminated areas inside the shield area, which houses the Main Turbine, Main Condenser, Moisture Separator Reheater, Feedwater Heaters and other significant heat loads. Outside the shield area includes the Main Generator, Air Compressor area, assorted electrical pumps and heat exchangers, and assorted electrical equipment. There are also two isolated Diesel Generator Rooms located on the Ground Floor, plus four enclosed stairwells. There is also a Battery Room on the Ground Floor. There are also two Switchgear Rooms located on the mezzanine level. Each is provided with chilled water supplied Fan Coil Units, and a small amount of supply air from two of the six TB AHUs for pressurization.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

The main generator is air-cooled and does not use hydrogen as the coolant. Therefore, there are no special requirements for hydrogen controls in the area.

The TB is supplied with outside air using six supply AHUs. These AHUs are provided with electric heating coils to provide a reasonable supply air temperature, and upstream prefilters to keep dirt off the coils, and for building cleanliness. The supply air is distributed through ductwork to areas both inside and outside the shield area. Most floors in the TB consists of metal grating, allowing supply air provided to lower elevations to increase in temperature and elevate naturally. Air is exhausted from the building via four AHUs with HEPA filtration located on the TB roof, taking suction from roof penetrations or suction ductwork. Air flow rates to areas have been chosen to effect transfer air from clean areas outside the shield area to potentially contaminated areas inside the shield area. TB supply and exhaust AHUs are provided AFDs to be able to reduce flow as permitted by plant operation and reduced ambient temperature conditions.

The PVS is located towards the north end of the TB roof. It is provided exhaust air from PVS exhaust fans venting the potentially radioactive air up and away from the TB. Each of the PVS exhaust fans takes suction on the Continuous Exhaust Air Plenum, which receives and mixes discharge air from the various building exhaust AHUs. The PVS radiation monitor is addressed in Chapter 11, Subsection 11.5.

The Diesel Generator Rooms are located in the TB. The Diesel Generator rooms are heated using electric unit heaters. The ventilation design of the Diesel Generator Rooms is for air to enter each room through two inlet air dampers connected to upstream ductwork and a common intake louver, providing outside air for both heat removal and for the combustion of diesel fuel. A small part of the exhaust air from the room exits from a wall mounted exhaust fan, located high on the wall. Most of the exhaust air from the rooms exits from the diesel generator as either radiator exhaust, ducted out the room through a wall-mounted louver/damper, or hard piped out the room through combustion piping/silencer.

The TB Battery Room is ventilated and conditioned by two mini-split HVAC systems, one for Train A and one for Train B, each system with an outdoor wall-mounted condenser unit and an indoor wall-mounted AHU, connected by refrigerant piping. Each AHU brings a small amount of filtered, conditioned outside air into the room, maintaining a slight positive pressure in the space. The room is heated by electric unit heaters.

The walls of the TB Battery Room and each of the two DG Rooms are constructed leak-tight relative to the TB to maintain radioactively clean environments.

#### **9A.5.4.3.1 Component Description**

Filters for the TB meet the applicable efficiency rating as stated below. ASHRAE Standard 52.2 (Reference 9A.5.4-5) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

Low Efficiency:	MERV 1-4 (Less than 20%)
Medium Efficiency:	MERV 5-12 (As least 20% but less than 80%)
High Efficiency:	MERV 13-16 (Greater than or equal to 80%)

HEPA (High Efficiency Particulate Air) Filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.4-5) are usually rated by the dioctylphthalate test method. This test is based on



the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle- collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.4-6).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.4-7).

### **Air Handler Units**

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.4-8).

### **Supply and Exhaust Fans**

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the applicable standards of AMCA (Air Movement and Control Association International). Fans are equipped with mechanisms to control airflows for the specific system application.

### **Heating Coils/Elements**

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (9A.5.4-6).

### **Cooling Coils**

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.4-9), and ANSI/ARI 410 (Reference 9A.5.4-10) and Underwriters Laboratory, (Reference 9A.5.4-11).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Subsection 9A.9.3) subsystem.

#### **9A.5.4.4 Materials**

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

#### **9A.5.4.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.4-2 for Turbine Building HVS interfaces with other equipment or systems

#### **9A.5.4.6 System and Equipment Operation**

##### **9A.5.4.6.1 Normal Operation**

The TB is normally provided heated, filtered once-through supply air from six normally operating AHUs located on the mezzanine level of the TB, which supply through ductwork to areas both inside and outside the radiological area. Air is exhausted from the same spaces through four normally operating exhaust AHUs located on the Turbine Building low bay roof, taking suction either directly through a TB roof penetration, or through suction ductwork connecting to the TB high bay. These exhaust AHUs discharge to the atmosphere from the PVS. AHU cooling provided by ventilation air is supplemented by cooling provided by FCUs located in areas that

require extra cooling, with FCUs being controlled by local thermostats. AHU heating is supplemented by electric unit heaters. Circulating fans are provided and used to mitigate hot spots within the building.

Exhaust air from potentially contaminated areas including the TB, RB, RWB, and Plant Service Area is filtered by AHUs using HEPA filters before being exhausted to the Continuous Exhaust Air Plenum. The Continuous Exhaust Air Plenum serves as a large mixing box where potentially contaminated air will be mixed and diluted. During normal operation, up to three PVS fans take suction on the Continuous Exhaust Air Plenum and discharge to the nearby PVS. The HEPA filters from the exhaust AHUs at each building assist in ensuring radioactive material entrained in gaseous effluent will not exceed the limits specified in SOR 2000-203 (Reference 9A.5.4-4), for normal operations and anticipated operational occurrences. The PVS fans are provided ASDs to be able to maintain the Continuous Exhaust Air Plenum negatively pressurized relative to the outside and to adjust to varying flow inputs, which may vary based on the operation of the exhaust AHUs at each building.

There are also two Switchgear Rooms. Each is provided with chilled water supplied FCUs, and a small amount of supply air from two of the six TB AHUs for pressurization.

Each enclosed stairwell is provided electric cabinet heaters to maintain space minimum temperature. These heaters are controlled by local thermostat.

Each of the two Diesel Generator Rooms located in the TB are provided two inlet air motor-controlled dampers, each with upstream ductwork leading to a common outside air intake louver. Each room has an exhaust fan discharging to the outdoors, and two electric unit heaters to maintain space design temperatures, controlled by local thermostat. An additional motor-controlled damper is designed to auto-open on Diesel Generator start and auto-close on DG stop with a time delay. This motor-controlled damper is provided in the Diesel Generator radiator cooling discharge ducted flow to the outdoors through a wall louver.

#### **9A.5.4.6.2 Off-Normal Operation**

In the event of an SBO, power is lost to all the TB HVAC equipment, Supply and exhaust AHUs and fan, electric unit heaters, FCUs, and all circulating fans.

In the event of a LOOP, with the exceptions of the Diesel Generator Room exhaust fans and motor-controlled dampers, two PVS Fans, and the operating Train A or B Battery Room mini-split cooling system, power is lost to all the TB HVAC equipment; Supply and exhaust AHUs, electric unit heaters, the FCUs, and all circulating fans.

A radiation monitor alarm in the TB initiates closure of the TB isolation dampers and secures the supply and exhaust AHUs. Refer to Chapter 11, Section 11.5 for information pertaining to radiation monitoring.

#### **9A.5.4.7 Instrumentation and Control**

The following describe key control features and the operational requirements for the TB HVS:

1. The TB HVS is operated from the Temperature Control Panel located near the Diesel Generator Rooms. TB Temperature Control Panel alarms are forwarded to the Main Control Room Panels.
2. TB supply AHUs are provided with Adjustable Speed Drives to ramp down air flow as outside air temperatures lower, prior to energizing integral electric heating coils.
3. AHU starting is not permitted until all associated isolation dampers are full open.

4. An AHU start request automatically opens all associated isolation dampers.
5. Failure of an operating damper automatically trips its associated AHU.
6. Failure of an operating AHU automatically closes all its associated isolation dampers.
7. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
8. TB FCUs are operated from TCPL.
9. TB Battery Room Train A and Train B mini-split systems are operated from TCPL.
10. TB Battery Room Train A and Train B mini-split systems are operated from the TB TCPL or from nearby touchpads.
11. Train B diesel generator room outside air inlet dampers and exhaust fans are operated from TCPL.
12. The Fire Protection System (Section 9A.6) provides a signal to shut down the turbine building AHUs in the event of a fire. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.
13. The supply AHUs are interlocked with the exhaust AHUs to ensure that the exhaust AHUs are running before a supply fan is started.

#### **9A.5.4.8 Monitoring, Inspection, Testing, and Maintenance**

The TB HVS provides isolation features to support testing and maintenance.

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.2-12).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

#### **9A.5.4.9 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.4.10 Performance and Safety Evaluation**

The TB HVS is not credited for mitigation of design basis accidents and not required to be operated during off-normal events.

HEPA filters associated with the TB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences. The exhaust air from the TB HVS is monitored for radioactivity prior to discharge to the plant vent. Alarms annunciate in the MCR upon detection of high radiation.

#### **9A.5.4.11 References**

9A.5.4-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.5.4-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.4-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.4-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."
- 9A.5.4-5 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.4-6 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.4-7 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.4-8 ASME AG-1 "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.
- 9A.5.4-9 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.4-10 AHRI Standard 410, "Forced Circulation, Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.4-11 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.4-12 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.5.4-1: Turbine Building, Heating, Ventilation and Air-Condition System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation monitoring of the HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs.	At the HVS equipment
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the TB HVS Non-Safety Category electrical loads.	At the equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning cooling coils.	At the air handler or fan coil unit flange
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides SC3 control to RB supply and exhaust isolation dampers, TB supply and exhaust dampers, and CRE EFUs.	At the TB HVS equipment
Safety-Category (TBD) Electrical Distribution System	Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs.	At the TB HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to TB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the TB HVS equipment

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

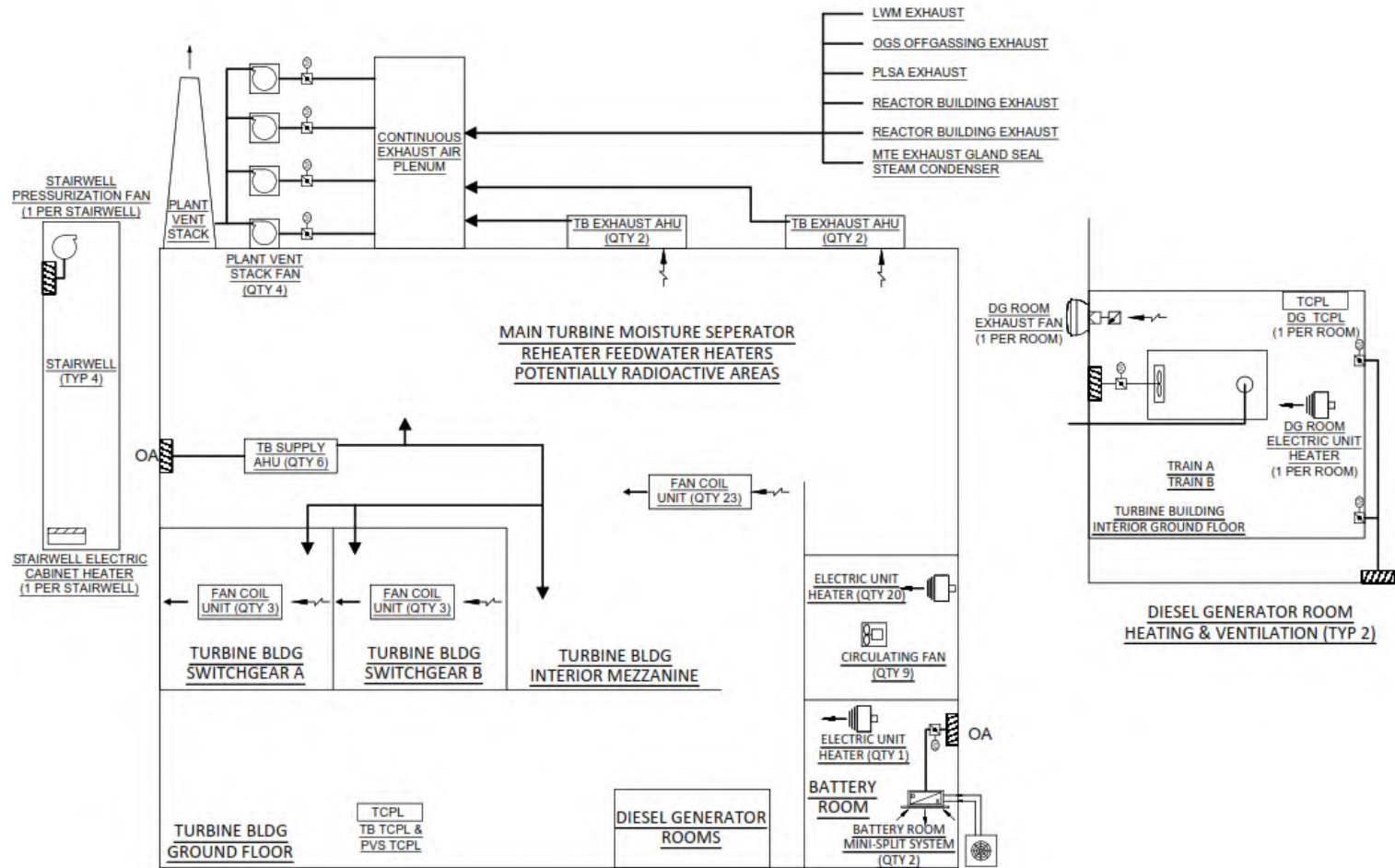


Figure 9A.5.4-1: Turbine Building HVAC

### **9A.5.5 Plant Services Area Heating, and Ventilation Systems**

The Plant Services Area is supplied by a 100% capacity makeup AHU and a 100% capacity exhaust AHU with HEPA filtration, functioning in a push-pull manner. Clean filtered outside air, heated when needed, is supplied through ductwork and supply registers to all spaces within the PLSAs. The exhaust AHU takes suction on these same spaces providing differential pressures resulting in flows from "clean" to "dirty" areas." AFDs are provided for each of the AHUs, permitting air flow to be ramped down in the wintertime, saving on heating load. Heating is supplemented by electric unit heaters located in most of the PLSA rooms. Baseboard heating units supplement the AHU heating in the Office and Auxiliary RCA Access Control Room. An electric unit heater is located near the exterior door of the hallway addressing the entryway into the North end of the building as well as along the various exterior walls of the rooms without baseboard heating.

Air discharged from the exhaust AHU is directed through ductwork to the Continuous Exhaust Air Plenum, where the air gets mixed with discharges from other building, before being exhausted out the PVS.

During recovery from a fire, smoke is exhausted from the PLSA by operating the supply AHU in 100% outside air mode in conjunction with operation of the exhaust AHU.

#### **9A.5.5.1 System and Equipment Functions**

##### **9A.5.5.1.1 Normal Functions (Non-Safety Category)**

1. The PLSA HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.5-1).
2. The PLSA HVS provides a controlled environment for the proper operation and integrity of equipment in the PLSA during normal, startup and shutdown operations.
3. The PLSA HVS provides necessary isolation features to support testing and maintenance.
4. The PLSA HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
5. The PLSA HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
6. The PLSA HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure.
7. The HVS reduces the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
8. The PLSA HVS is designed such that failure of SCN equipment does not compromise or otherwise damage Safety-Class equipment.
9. The PLSA HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

#### **9A.5.5.1.2 Normal Functions (Safety-Category)**

The PLSA HVS does not perform Safety-Category functions.

#### **9A.5.5.1.3 Off-Normal Functions (Non-Safety-Category)**

PLSA HVS continue to operate during off-normal conditions until removed from service either manually or automatically.

#### **9A.5.5.1.4 Off-Normal Functions (Safety-Category)**

The PLSA HVS does not perform any Safety-Category functions.

The PLSA HVS is not required to operate during or after a design basis event. The design of the PLSA HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.5-2) as related to providing a ventilation system with an appropriate filtration system capable of the following:

1. Preventing unacceptable dispersion of all airborne contaminants within the plant through the use of the exhaust AHUs to maintain a negative pressure in clean areas, resulting in air transferring from clean to "dirty" areas."
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the PLSA HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

#### **9A.5.5.2 Safety Design Bases**

The PLSA HVS does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

#### **9A.5.5.3 Description**

Figure 9A.5.5-1 depicts the PLSA HVAC Process Flow Diagram.

The PLSA are provided filtered, heated air from a normally operating supply AHU located on the Plant Services Area roof, supplying through ductwork. Air is exhausted from the spaces by a normally operating exhaust AHU with HEPA filtration also located on the roof, taking suction through ductwork, discharging ultimately to the outdoors through the PVS. Heating is supplemented by electric unit heaters and baseboard heaters.



The PLSA Temperature Control Panel is provided in the Plant Services Area Office and is used to control the supply and exhaust AHU.

#### **9A.5.5.3.1 Component Description**

##### **Air Handling Units**

Each AHU consists of an inlet area, filters, heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.5-5).

##### **Supply and Exhaust Fans**

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of AMCA. Fans are equipped with ASD mechanisms as required to control airflows for the specific system application.

##### **Heating Coils/Elements**

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (Reference 9A.5.5-6).

#### **9A.5.5.4 Materials**

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

#### **9A.5.5.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.5-1 for PLSA HVS interfaces with other equipment or systems.

#### **9A.5.5.6 System and Equipment Operation**

##### **9A.5.5.6.1 Normal Operation**

The PLSA is provided filtered, heated air from a normally operating supply AHU located on the PLSA roof, supplying through ductwork. Air is exhausted from the space by a normally operating exhaust AHU also located on the roof, taking suction through ductwork, discharging ultimately to the outdoors through the PVS. Heating in each of the ventilated rooms is supplemented by electric unit heaters or baseboard heaters located in each room. The PSA supply AHU and exhaust AHUs are controlled at the Temperature Control Panel in the PLSA Office. The unit heaters are all controlled by local thermostats.

##### **9A.5.5.6.2 Off-Normal Operation**

In the event of a LOOP or SBO, power is lost to the Plant Services Area HVAC equipment including the supply and exhaust AHUs, electric unit heaters, and baseboard heaters.

#### **9A.5.5.7 Instrumentation and Control**

Manual initiation and shutdown of the PLSA AHUs is provided from the PLSA temperature control panel located in the Plant Services Area Office. System and component operating status for the AHUs is displayed in the MCR:

1. Manual initiation and shutdown of the Plant Services Area AHUs is provided from the Plant Services Area Temperature Control Panel located in the Plant Services Area Office.

2. An AHU start request automatically opens all associated isolation dampers.
3. Failure of an operating damper automatically trips its associated AHU.
4. Failure of an operating AHU automatically closes all its associated isolation dampers.
5. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
6. Electric Unit Heaters are controlled from local wall-mounted thermostats.
7. The FPS provides a signal to shut down the Plant Services Area AHUs in the event of a fire. Duct mounted smoke detectors provide a smoke detection signal to the Fire Protection System local fire protection panels.

#### **9A.5.5.8 Monitoring, Inspection, Testing, and Maintenance**

The PLSA HVS provides necessary isolation features to support testing and maintenance

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.5-7).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

#### **9A.5.5.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.5.10 Performance and Safety Evaluation**

The PLSA HVS does not perform any Safety-Category functions and therefore requires no nuclear safety evaluation.

The PLSA HVS is not credited for mitigation of design basis accidents and not required to be operated during off-normal events.

HEPA filters associated with the PLSA HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences. The exhaust air from the PLSA HVS is monitored for radioactivity prior to discharge to the plant vent. Alarms annunciate in the MCR upon detection of high radiation.

#### **9A.5.5.11 References**

- 9A.5.5-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."
- 9A.5.5-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.5-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.5-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.5.5-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.
- 9A.5.5-6 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.5-7 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

**Table 9A.5.5-1: Plant Services Area Heating, and Ventilation System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads.	At the HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to TB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the HVS equipment

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

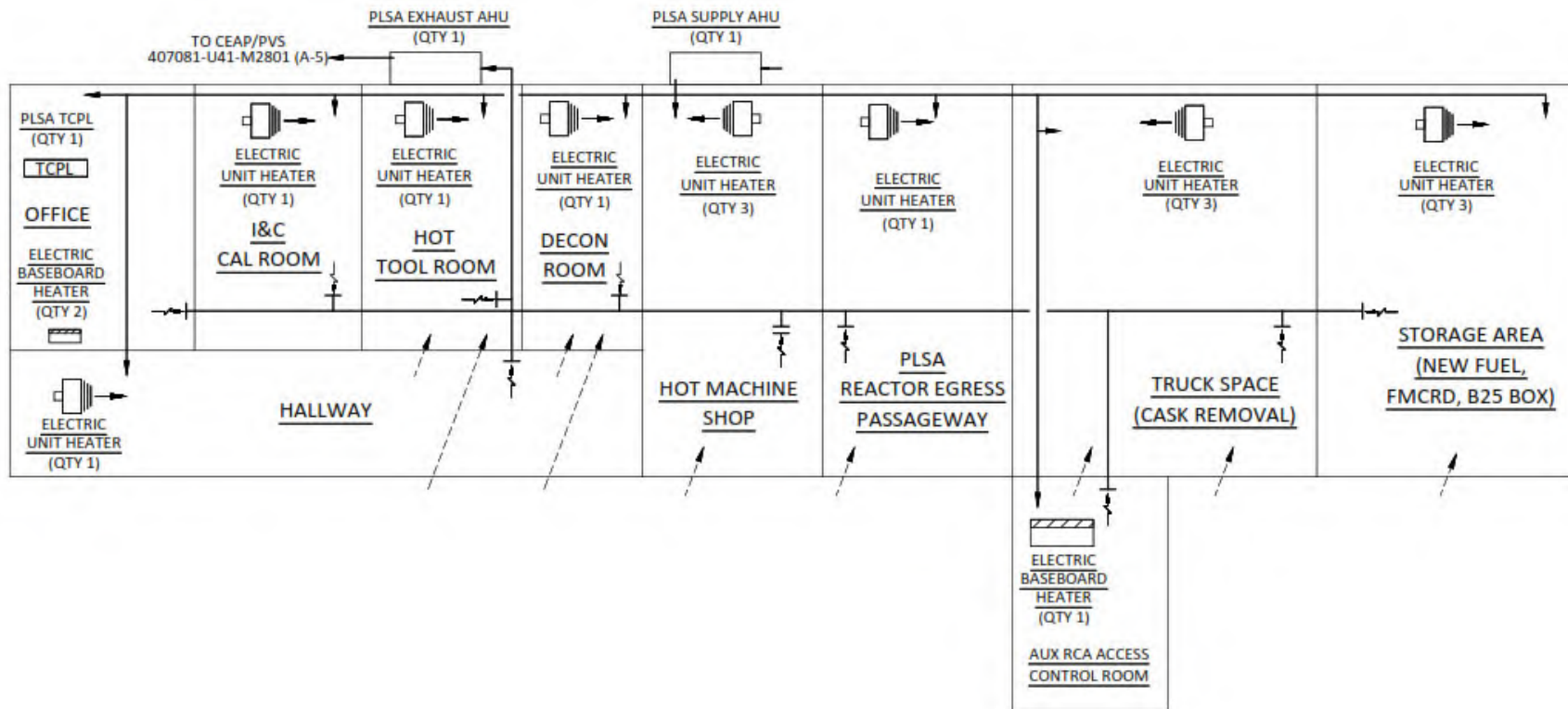


Figure 9A.5.5-1: Plant Services Area HVAC

## **9A.5.6 Containment Cooling System**

The Containment Cooling System is a closed loop cooling system that recirculates air/nitrogen with no outside air introduced into the system except during outages.

### **9A.5.6.1 System and Equipment Functions**

The CCS supports performance of the following functions during operating conditions through the entire operating range, from startup to full load condition to refueling:

#### **9A.5.6.1.1 Normal Functions (Non-Safety-Category)**

The CCS performs the following Non-Safety-Category functions during normal operations.

- CCS aids in purging of nitrogen from the containment during shutdown

#### **9A.5.6.1.2 Normal Functions (Safety-Category)**

The CCS performs the following Safety-Category functions during normal operations:

- CCS maintains temperature in the upper containment and control rod drive area at the required target temperature and below the specified maximum temperature limit during normal operation
- CCS maintains operation temperature envelope in the Steel-plate Composite Containment Vessel (SCCV) to support Environmental Qualification (EQ) of Safety-Category function related equipment during Modes 1-5

#### **9A.5.6.1.3 Off-Normal Functions (Non-Safety-Category)**

The CCS performs the following Non-Safety-Category functions during Off-Normal operations:

- CCS supports heat removal during outages, if available

#### **9A.5.6.1.4 Off-Normal Functions (Safety-Category)**

The CCS performs the following SC3 Safety-Category functions during off-normal operations:

- CCS assists with containment cooldown following a Loss-of-Offsite Power (LOOP) during the period from hot shutdown to cold shutdown if available
- CCS limits containment temperature during a LOOP.

### **9A.5.6.2 Safety Design Bases**

The CCS is classified as Defense Line 2 and Safety Class 3 (DL2/SC3). The CCS performs the Safety-Category function of heat removal in the containment.

### **9A.5.6.3 Description**

Figure 9A.5.6-1 depicts the CCS.

The design of the CCS provides cooling to the containment consistent with the requirements of REGDOC 2.5.2 Section 7.10 (Reference 9A.5.6-1).

The CCS is Safety Class 3. The CCS is used to ensure the containment conditions are maintained, which includes the EQ temperatures envelope for Safety-Category function related equipment.

The CCS provides containment cooling using AHU that reject heat to the CWE (Subsection 9A.2.4) during all modes of operation. The CCS is a closed loop recirculating cooling system.

The CCS is comprised of four (4) 50% capacity AHUs and is divided into two trains of two (2) AHUs. Each train is cooled by a corresponding chilled water train so that SCCV cooling is still possible even with loss of one train of CWE or one train of CCS. Each AHU can be powered from one of the standby diesel generators of the Safety Class 2 and 3 Electrical Distribution System.

During normal operation, only one train of CCS is operating while the other is in standby. The train in standby automatically starts upon loss of the lead train or upon temperature exceeding target. All AHUs can be in service at the same time, if needed.

Each AHU has a condensate collection pan. The condensate collected from the AHUs is drained to a hub location inside of containment, then is piped to the Equipment and Floor Drain System (Subsection 9A.9.3) sump located outside containment where it is discharged.

The CCS air distribution ductwork is constructed of hot-dipped galvanized sheet metal and uses manual balancing dampers to control the distribution of air to the various containment locations.

Backdraft dampers are provided on each fan suction to prevent reverse rotation of a non-operating fan.

The system is classified as Seismic Interaction which requires evaluation to ensure that during a seismic event there are no adverse interactions with the ability of SSCs classified as Seismic A or B to perform their functions. Refer to Chapter 3, Subsection 3.2 for additional information pertaining to Seismic Interaction.

The CCS ductwork and equipment are designed in accordance with ASHRAE (Reference 9A.5.6-2) for the reactor building environment to support the operation and integrity of equipment during normal, startup and shutdown operations.

#### **9A.5.6.3.1 Component Description**

##### **Air Handling Units**

The CCS is comprised of two trains of AHUs. Each AHU is cooled by the corresponding CWE system train. The AHU cooling coils are to be designed, constructed, and installed in manner that facilitates coil cleaning and supports ease of replacement as required. A manual damper is installed at the AHU suction for isolation during maintenance.

Low efficiency air filter sections are provided upstream of the cooling coil sections to permit installation of temporary filters during testing and outages.

##### **Supply Fans**

There are four (4) 50% capacity recirculating supply air fans in CCS. The supply fans are located downstream of their respective AHU coils. A totally enclosed fan cooled motor is provided for the fan.

##### **Cooling Coils**

CCS has four (4) cooling coils per train. Supply fans are located downstream of the cooling coils. These coils use CWE chilled water as the heat sink.

Condensate from the cooling coils is routed to the bottom of the containment, then through piping to the EFS sump located outside of containment. All cooling coils are provided with stainless steel condensate drain pans.

##### **Air Filters (Temporary)**

Low efficiency air filter sections are provided upstream of the cooling coil sections. Low efficiency air filters are used during construction and post-construction testing of CCS to protect the cooling

coils. After the above test, the air filters are removed since it is not necessary to filter the recirculating air during normal plant operation. If necessary, during a refueling outage, temporary air filters can be installed in the filter section.

### **Piping, Valves, and Ductwork**

CCS piping and valves are classified as follows:

CCS cooling coil condensate drain piping is SCN and designed to meet ASME B31.1 Power Piping requirements.

CCS ductwork and dampers are classified as follows:

CCS ductwork and dampers are designed, installed and inspected according to ASHRAE Fundamentals, SMACNA, HVAC System Duct Design, and SMACNA Duct Construction Standards. The ductwork is seismically supported so that in the event of a failure of CCS Structures, Systems, or Components during a seismic event does not adversely affect the ability of any Seismic Category A or B Structure, System or Component to accomplish its Safety Class function.

#### **9A.5.6.4 Materials**

Selection of radiation-resistant materials of construction is included in individual equipment specifications.

Material and equipment selection for CCS components is based on a 60-year plant life, with appropriate provisions for maintenance and replacement. Components which may require periodic replacement and/or maintenance prior to the end of plant life are dampers, damper linkages, AHUs, bearings, and motors.

Aluminum ducting is not allowed inside containment.

Materials are selected in accordance with applicable codes, standards, and industry practice for the design, service, and test conditions and expected ambient conditions.

Materials selected are compatible with the internal process and external environmental conditions during normal, abnormal, accident, and beyond design basis accident conditions as appropriate.

#### **9A.5.6.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.5.6-2 for CCS interfaces with other equipment or systems.

#### **9A.5.6.6 System and Equipment Operation**

CCS is designed for continuous operation for the following operating modes.

##### **Normal Operational Concept**

CCS normal plant operation includes startup, power operation, hot shutdown, cold shutdown, and refueling. In normal operation mode, one (1) of the two (2) trains operates as lead while the other train is in standby. The lead train is initiated remotely from the Main Control Room. The standby train is automatically started when containment temperature reaches limit or upon tripping of the lead train to maintain the normal operation temperature envelope and support environmental qualification of Safety-Category function related equipment. The assignment for lead and standby train is switched periodically. Both trains can also operate simultaneously.

##### **Off-Normal Operational Concept**

The CCS AHUs and supporting chillers are provided with backup power from the diesel generators. The CCS remains operational during a Loss-of-Offsite Power. This supports



maintaining the containment environment from exceeding design conditions. The CCS assists with containment cooldown following a Loss-of-Offsite Power during the period from hot shutdown to cold shutdown.

#### **9A.5.6.7 Instrumentation and Control**

CCS provides the instrumentation to control and monitor system operation.

Operation of the CCS is controlled remotely from the Main Control Room. Each AHU has a local disconnect switch to facilitate maintenance. Temperature elements are provided to indicate and alarm on loss of train or temperature over target and energize the standby fan motor. Containment electrical penetrations are used to route the cable from the control elements in the ductwork to the Main Control Panel. There will be no CCS local control panels.

#### **9A.5.6.8 Monitoring, Inspection, Testing, and Maintenance**

The cooling coils can be accessed for cleaning. Only water can be used for cleaning. No compressed air is allowed. There is no need for duct cleaning since no outside air is introduced during operation.

Pick points for heavy equipment are provided. The equipment and components of the CCS are designed for inspection and maintenance accessibility. Additionally, fans, motors, coils, filter section, and dampers and damper actuators can be removed for maintenance and repair.

To minimize maintenance (e.g., changing/adjusting belts) and contribute to ALARA goals, the CCS fans are equipped with a quick disconnect on the motor leads and a motor terminal junction box. The fans are provided with full manual control for testing and maintenance. Ductwork is arranged to facilitate AHU maintenance.

To allow testing and balancing, small holes may be drilled on the side of the CCS ductwork to take readings with temporary instrumentation. The holes are capped after the testing and balancing is completed.

AHUs are inspected and maintained according to the manufacturer's requirements.

CCS piping is installed and inspected according to ASME B31.1. CCS ductwork is installed and inspected according to Sheet Metal and Air Conditioning Contractors National Association, Inc.

#### **9A.5.6.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to ALARA design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.5.6.10 Performance and Safety Evaluation**

CCS equipment is classified as Safety Class 3 and seismic classification "Seismic Interaction" which entails an evaluation as discussed in Chapter 3, Subsection 3.2 to ensure that in the event CCS Structures, Systems and Components, fail during a seismic event there are no adverse interactions with the ability of any Seismic Category A or B Structure, System, or Component to accomplish their Safety-Class function.

The CCS is designed with N+1 redundancy for asset protection. In the event of one train failure or shut down for maintenance, the train in standby automatically starts and maintains full operation of the system to protect the EQ equipment inside the SCCV.

**9A.5.6.11 References**

- 9A.5.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.6-2 "ASHRAE Handbook– Fundamentals," ASHRAE.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.5.6-1: Containment Cooling System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Safety Class 2 and 3 Instrumentation and Control	Provides all Safety Class 3 I&C control, logics, and instrumentation (pressure, flow, temperature), etc.	AHUs and instruments
Chilled Water Equipment System	CCS transfers heat to the CWE	Cooling Coils flanges
Safety Class 2 and 3 Electrical Distribution System	Provides power to AHUs	AHUs and instruments
Steel-plate Composite Containment Vessel	Contains environment provided by CCS, thereby ensuring that the environmental qualification limits of the Safety-Category function related equipment inside of the SCCV are not exceeded	Safety-Category function related equipment
Equipment and Floor Drain System	The condensate collected from the AHUs drain pans is routed to the EFS sump located outside of the containment	Drain pipe

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

ABBREVIATIONS:

CCS - CONTAINMENT COOLING SYSTEM  
CWE - CHILLED WATER EQUIPMENT SYSTEM  
EFS - EQUIPMENT AND FLOOR DRAIN SYSTEM  
SA - SUPPLY AIR  
RA - RETURN AIR

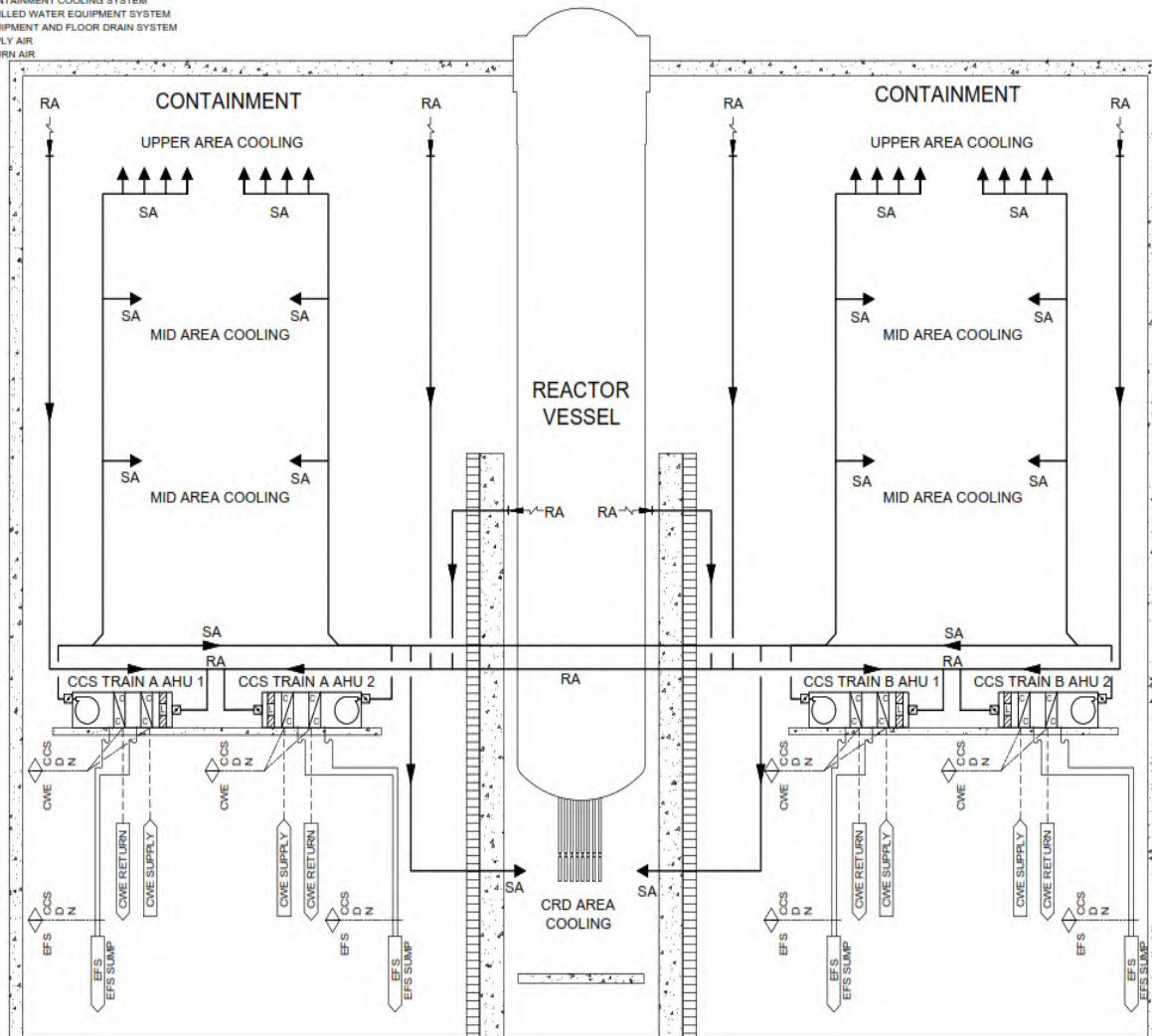


Figure 9A.5.6-1: Containment Cooling System

## **9A.6 Fire Protection Systems**

### **Introduction**

This section provides information on the BWRX-300 fire protection system design. It also confirms that the design meets the Canadian Nuclear Safety Commission (CNSC) expectations for fire protection system design.

The BWRX-300 fire protection system is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires.

This document addresses the requirements of CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants (Reference 9A.6.11-3), by providing a set of comprehensive design requirements and guidance that is risk-informed to align with accepted international codes and practices.

The guidance in this document is consistent with international best practices and the recommendations of the International Atomic Energy Agency and International Commission on Radiological Protection.

This section provides criteria pertaining to the safe design of the BWRX-300 in Canada. Aspects of the BWRX-300 design are considered, and multiple levels of defence are promoted in design considerations.

A summary description of the BWRX-300 fire protection system design is provided below. In addition, more detailed information describing the salient features of the overall BWRX-300 Fire Protection Program (FPP).

This discussion focuses on fire protection design consisting of the following:

- General design approach and strategy for fire protection
- Structural aspects of fire protection
- Description of fire protection measures inside confinement and containment
- Strategy and measures for alerting staff of fire events or conditions that may potentially trigger a fire event
- Strategy and measures for control of fire protection systems
- Human factors considerations in the design for fire protection, and
- Description of fire protection systems interface with other systems, including inter-unit interfaces, where common systems are shared including these systems which have a Safety Class classification.

### **Fire Protection Program General Information**

The BWRX-300 fire protection system is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires. The Fire Protection Program is different than the Fire Protection System (FPS) which is a system of mechanical equipment and piping used to provide fire detection and suppression to fight a fire at the plant.

The OPG DNNP Fire Protection Program includes the concepts of design and layout implemented to prevent or mitigate fires, administrative controls and procedures, and the training of personnel to combat fires. Refer to Chapter 3, Subsection 3.4.1 for a discussion of the Fire Protection design

requirements and features for SSCs for the BWRX-300. Additional discussion of the Fire Protection Program is provided in Chapter 19, Section 19.5.

The fire protection program design addresses protection from fire by demonstrating that a Defence-in-Depth (D-in-D) approach has been implemented. The Fire Protection Assessment ensures implementation of the fire protection program through delivery of supporting documents including a comprehensive design report, code compliance review, a fire hazard assessment, and fire safe shutdown analysis.

The PSAR includes results of an independent third-party review of the design, assessing compliance against the applicable fire codes and standards used in the design for protection from fires and explosions. Refer to Section 9A.6.10 for a discussion of the third-party review.

#### **9A.6.1 System and Equipment Functions**

The FPS performs the following functions during normal and off-normal conditions.

##### **Normal Functions (Non-Safety Category)**

FPS Provides the following safety class functions:

1. Detect and locate fires by fire area or fire zone and actuate alarms promptly (excluding the inerted Primary Containment).
2. Provide automatic and manual fire suppression systems, where required by the Fire Hazard Assessment (FHA) of the BWRX-300 as well as in those areas as defined by other future life safety and fire task analysis.
3. Provide manual backup to automatic fire suppression systems.
4. Limit the spread of fires.
5. Suppress fires which do occur, thereby minimizing the adverse effects of fire on structures, systems, and components important to safety and to production of electricity and economic loss.
6. Ensure that one of the redundant divisions necessary to achieve safe shutdown is free of fire damage.
7. Minimize radioactive or hazardous chemical exposure to personnel and radioactive or hazardous chemical release to the environment as the result of a fire.
8. Provide firewater supply and distribution of firewater to all suppression circuits.

##### **Normal Functions (Safety Category)**

The FPS does not perform any safety class functions.

##### **Off-Normal Functions (Non-Safety Category)**

The FPS continues to perform its functions in off-normal conditions including a Loss-of-Offsite Power (LOOP). Electronic elements of the system have their own batteries or are electrically supplied from the uninterruptible power supply portion of the electrical system (Refer to Chapter 8).

##### **Off-Normal Functions (Safety Category)**

The FPS does not perform any Off-Normal safety category functions.

## **9A.6.2 Safety Design Bases**

### **Fire Protection System Objective**

The BWRX-300 Fire Protection System (FPS) is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires.

### **Safety Design Bases**

The FPS does not perform any SC1 or SC2 functions.

### **Power Generation Design Bases**

In accordance with CSA N293, Fire protection for nuclear power plants and N293S1, Supplement No. 1 to N293 (References 9A.6.11-1 and 9A.6.11-7), Fire protection for nuclear power plants (application to small modular reactors) the FPS uses the concept of D-in-D to achieve the required degree of reactor safety by using administrative controls, fire protection systems and features, and safe shutdown capability. These D-in-D principles achieve the following objectives:

- Prevent fires from starting:
  1. And rapidly detect, control, and extinguish promptly those fires that do occur
  2. To provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant and does not significantly increase the risk of radioactive release to the environment.
- Provide timely identification and suppression of fires:
  1. To provide automatic fire detection and annunciation for selected areas of the plant as required by the FHA for personnel safety and fire brigade notification.
  2. To supply the maximum firewater demand at any point throughout the system, with one fire pump out of service.
  3. To maintain the ability to safely shut down the reactor and keep it shut down by providing the capability to control the spread of and extinguish the postulated fires in all plant areas by the use of fixed and/or portable firefighting equipment. This capability is achievable during all modes of plant operation.
  4. To prevent inadvertent operation of the FPS from jeopardizing the capability to achieve safe shutdown of the plant.
  5. To ensure a continuous firewater supply for the fire pumps in the event of failure of one firewater source. Two separate firewater sources are connected to FPS such that there is no interruption in supply and that failure of one water source or its piping does not drain the other sources.
  6. To provide manual suppression capability to all plant areas, including those that have automatic fire suppression systems.
  7. To ensure at least one effective hose stream can reach any location containing SC1 or SC2 equipment, for preventing a fire exposure hazard to the equipment.
  8. To permit isolation from the fire main or outside hydrants for maintenance or repair without interrupting FPS water supply.

- Ensure that the plant can be safely shutdown in the event of a fire:
  1. To maintain the ability to safely shut down the reactor and keep it shut down by providing adequate separation of SC1 or SC2 equipment. This capability is to be achievable during all modes of plant operation.
  2. To minimize the probability of the spread of fire by the use of fire barriers between areas of significant combustible loading.
  3. To maintain the ability to minimize the potential for radioactive releases to the environment in the event of a fire.
  4. To preclude damage to plant SC1 or SC2 structures, systems, or components caused by seismic loading of the FPS.
  5. To keep equipment required for safe shutdown free from fire damage during a Safe Shutdown Earthquake (SSE). To this end, one source of firewater supply, including a water source, two fire pumps and their associated suction and discharge lines; and firewater lines, including standpipes and hose connections, are designed, and analyzed to remain functional after an SSE. This includes analysis to the first isolation valve on all branches connected to the seismically analyzed firewater lines.
  6. To have a useful life of 60 years with normal maintenance and replacement of parts/components subject to normal wear and deterioration.

#### **Fire Protection System Design Codes and Standards**

Mechanical System Codes and Standards utilized for conceptual design are listed in PSAR Chapter 1. A subset of this list addresses relevant FP design codes and standards. The applicable BWRX-300 FP design codes and standards are provided Table 9A.6.2-1.

Chapter 9A, Table 9A.6.2-1 also provides a list of key documents applicable to BWRX-300 FPS design.

A preliminary Code Compliance Review has been performed as part of the FPP development and list the codes and standards required for BWRX-300 FP system design and program development. This document defines code compliance or lists open items, exemptions or alternatives that depart from Canadian codes and standards and how the intent of the requirements is met using equivalent or alternative means.

This Fire Hazard Assessment (FHA) (Reference 9A.6.11-5) establishes and evaluates distinct fire areas for the Reactor Building, Radwaste Building, Turbine Building, Control Building, PLSA and Fire Pump Enclosure for OPG DNNP BWRX-300 plant.



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.6.2-1: Applicable BWRX-300 Fire Protection Design Codes and Standards and Key Documents**

Code and Standard	Title
<b>American Society of Mechanical Engineers (ASME)</b>	
B31.1	Power Piping
<b>American Society for Testing and Materials</b>	
ASTM D323	Standard Test Method for Vapor Pressure of Petroleum Products (Reid Method)
<b>CSA Group</b>	
CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code
CSA B72	Installation code for lightning protection systems
CSA C22.1 Canadian Electrical Code, Part I	Safety Standard for Electrical Installations
CAN/CSA-C22.2 NO. 0.17	Evaluation of Properties of Polymeric Materials
CSA C22.2 NO. 2556	Wire and cable test methods (Trinational standard with NMX-J-556-ANCE-2021 and UL 2556)
CSA C282	Emergency electrical power supply for buildings
CSA N289.3	Design procedures for seismic qualification of nuclear power plants
CSA N293	Fire protection for nuclear power plants
CSA N293 Supplement 1	Fire protection for nuclear power plants (Application to small modular reactors)
<b>Electric Power Research Institute (EPRI)</b>	
EPRI Product ID 1006756	Technical Report Fire Protection Equipment Surveillance Optimization and Maintenance Guide
EPRI NP-2660	Fire Tests in Ventilated Rooms, Extinguishment of Fire in Grouped Cable Trays,
<b>Factory Mutual</b>	
Factory Mutual 7-101	Fire Protection for Steam Turbines and Electric Generators

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Code and Standard	Title
<b>International Atomic Energy Agency (IAEA)</b>	
International Atomic Energy Agency, IAEA Safety Standards Series, Safety Guide No. NS-G-2.1	Fire Safety in the Operation of Nuclear Power Plants
IAEA, Safety Report Series No. 8	Preparation of Fire Hazard Analysis for Nuclear Power Plants
IAEA, NS-G-1.7	Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants
INSAG Series No. 10	Defence-in-Depth in Nuclear Safety
INSAG Series No. 12	Basic Safety Principles for Nuclear Power Plants 75-INSAG-3
Safety Reports Series No. 10	Treatment of Internal Fires in Probabilistic Safety Assessment for Nuclear Power Plants
Safety Reports Series No. 46	Assessment of Defence-in-Depth for Nuclear Power Plants
IAEA Safety Standards Series No. NS-G-1.7	Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants
IAEA Safety Standards Series No. NS-G-2.1	Fire Safety in the Operation of Nuclear Power Plants
<b>National Research Council</b>	
NBCC	National Research Council National Building Code of Canada
NFCC	National Research Council National Fire Code of Canada
<b>Nuclear Energy Institute (NEI)</b>	
NEI 00-01	Guidance for Post Fire Safe Shutdown Circuit Analysis, NEI Circuit Failure Issues Task Force
<b>National Fire Protection Association (NFPA)</b>	
NFPA 1	Uniform Fire Code Handbook
NFPA 10	Standard for Portable Fire Extinguishers
NFPA 11	Standard for Low-, Medium- and High-Expansion Foam Systems
NFPA 13	Standard for the Installation of Sprinkler Systems

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Code and Standard	Title
NFPA 13E	Fire Department Operations in Properties Protected by Sprinkler and Standpipe Systems
NFPA 14	Standard for the Installation of Standpipe and Hose Systems
NFPA 15	Standard for Water Spray Fixed Systems for Fire Protection
NFPA 16	Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems
NFPA 17	Standard for Dry Chemical Extinguishing Systems
NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection
NFPA 22	Standard for Water Tanks for Private Fire Protection
NFPA 24	Standard for the Installation of Private Fire Service Mains and their Appurtenances
NFPA 25	Standard for Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems
NFPA 30	Flammable and Combustible Liquids Code
NFPA 30B	Code for Manufacture and Storage of Aerosol Products
NFPA 55	Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks
NFPA 69	Standard on Explosion-Prevention Systems
NFPA 72	National Fire Alarm and Signaling Code
NFPA 76	Standard for the Fire Protection of Telecommunications Facilities
NFPA 80A	Recommended Practice for Protection of Buildings from Exterior Fire Exposures
NFPA 90A	Standard for the Installation of Air Conditioning and Ventilating Systems
NFPA 90B	Standard for the Installation of Warm Air-Heating and Air Conditioning Systems
NFPA 91	Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists and Noncombustible Particulate Solids

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Code and Standard	Title
NFPA 92	Standard for Smoke-Control Systems
NFPA 92A	Recommended Practice for Smoke-Control Systems
NFPA 101	Life Safety Code
NFPA 241	Standard for Safeguarding Construction, Alteration, and Demolition Operations
NFPA 600	Standard on Industrial Fire Brigades
NFPA 701	Standard Methods of Fire Tests for Flame Propagation of Textiles and Films
NFPA 780	Standard for the Installation of Lightning Protection Systems
NFPA 801	Standard for Fire Protection Practices for Facilities Handling Radioactive Materials
NFPA 804	Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants
NFPA 805	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants
NFPA 850	Recommended Practice for Fire Protection for Electric Generating Plants and High-Voltage Direct Current Converter Stations
NFPA 855	Standard for the Installation of Stationary Energy Storage Systems
NFPA 921	Guide for Fire and Explosion Investigations
NFPA 1081	Standard for Industrial Fire Brigade Member Professional Qualifications
NFPA 1620	Recommended Practice for Pre-Incident Planning
NFPA 2001	Standard on Clean Agent Fire Extinguishing Systems
National Fire Protection Association (NFPA)	Fire Protection Handbook
SFPE	Society of Fire Protection Engineers Engineering Guide to Performance-based Fire Protection
Nuclear Insurance Pools Forum (NIPF)	International Guidelines for the Fire Protection of Nuclear Power Plants

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Code and Standard	Title
<b>Ontario Code</b>	
Ontario Building Code	Ontario Building Code
Ontario Fire Code	Ontario Fire Code
<b>Underwriters Laboratory (UL)</b>	
UL 586	UL Standard for Safety High Efficiency, Particulate, Air Filter Units
ULC CAN/ULC-S102	Method of Test for Surface Burning Characteristics of Building Materials and Assemblies
ULC CAN/ULC-S102.2	Method of Test for Surface Burning Characteristics of Flooring, Floor Covering and Miscellaneous
ULC CAN/ULC-S107	Method of Fire Tests of Roof Coverings
ULC CAN/ULC-S109	Flame Tests of Flame Resistant Fabrics and Films
ULC CAN/ULC-S111	Standard Methods of Fire Tests for Air Filter Units
ULC CAN/ULC-S114	Standard Method of Test for Determination of Non-Combustibility in Building Materials
ULC CAN/ULC-S115	Standard Methods of Fire Tests for Fire Stop Systems
ULC CAN/ULC-S126	Standard Method of Test for Fire Spread Under Roof-Deck Assemblies
ULC CAN/ULC-S524	Standard for the Installation of Fire Alarm Systems
ULC CAN/ULC-S537	Verification of Fire Alarm Systems
<b>U.S. Nuclear Regulatory Commission (NRC)</b>	
NRC NUREG-1852	Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire
U.S. NRC, NUREG-0800, Section 9.5.1.1	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR edition - Fire Protection Program
U.S. NRC, Regulatory Guide 1.189	Fire Protection for Operating Nuclear Power Plants

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.6.2-2: System Interfaces**

Name	Interfacing	Interface Description	Interface Boundary
C20	Safety Class 2 and 3 Instrumentation and Control System	C20 provides the instrumentation signals for the system to the fire panels in the MCR/SCR and in areas where safety class 2 and 3 instrumentation is required.	In the MCR/SCR
C30	Non safety class Distributed Control and Instrumentation System (SCN- DCIS)	C30 receives instrumentation signals from the FPS through the fire panels in the MCR/SCR.	In the MCR/SCR
H11	Main Control Room (MCR) Panels	The MCR Panels provides a remote manual actuation capability for the U43 FPS pumps and valves.	In the MCR
K30	Offgas System	The FPS provides firewater for the manual spray on the charcoal absorbers.	In the Radwaste building at each charcoal absorber
P52	Plant Pneumatics System (PPS)	PPS supplies dry compressed air to the preaction and dry pipe sprinkler systems.	At dry pipe and preaction FPS sprinklers
R20	Safety Class 2 and 3 Electrical Distribution System	R20 provides diesel backed electrical power to FPS equipment.	At the Fire Pump Enclosure
R30	Non-safety class Electrical Distribution System	R30 provides non-safety power to FPS equipment.	At the Fire Pump Enclosure
U31	Load Handling System (LHS)	The FPS supervises fire detection in elevator lobbies, shafts, and machine rooms and sends signals to elevator controllers for recall and to shunt trip breakers for power shunt.	At the Local Fire Control Panels nearest to each elevator
U41	Heating, Ventilation	The Plantwide HVS removes smoke from fire areas. Duct- mounted smoke detectors transmit an alarm signal to the local fire control panel upon detection of smoke. Fire/Smoke dampers are considered U41 SSC.	Plantwide, having multiple interfaces in each building for the U41 HVAC system
U50	Equipment and Floor Drain System (EFS)	EFS collects fire protection water during FPS testing and actuation.	Located in the lowest levels of the turbine, radwaste, and reactor building

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Name	Interfacing	Interface Description	Interface Boundary
Y53	Water, Gas, and Chemical Pads (WGC)	Onsite Fuel Storage provides supply for refilling the diesel pump day tanks.	Later
Y99	BWRX Yard	Provides municipality water supply for the firewater make- up to the firewater storage tanks.	Located at the border of the protected area

### **9A.6.3 Description**

#### **Fire Safety System Design Objectives**

The primary objectives of BWRX-300 FPS are to minimize both the probability of occurrence and the consequences of fire. To meet these objectives, the FPP is designed to provide reasonable assurance, through D-in-D, that a fire does not prevent the necessary safe shutdown functions from being performed and that radioactive releases to the environment in the event of a fire is minimized.

#### **Development of Fire Protection Performance Criteria and Assessment Metrics**

##### **Nuclear Safety Objectives**

In the event of a fire, the plant is capable of:

- Achieving and maintaining the reactor in subcritical conditions
- Achieving and maintaining decay heat removal
- Maintaining the integrity of the fission product boundaries
- Limiting the release of radioactive materials

### **9A.6.4 Materials**

Noncombustible and heat-resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment and control rooms. Fire detection and firefighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on SSCs important to safety. The firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

### **9A.6.5 Interfaces with Other Equipment or Systems**

#### **Fire Protection System Interdependencies with Other Systems**

Fire Protection interfaces are listed in PSAR Table, System Interfaces (Chapter 9A, Table 9A.6.2-2)



### **9A.6.6 Fire Protection System and Equipment Operation**

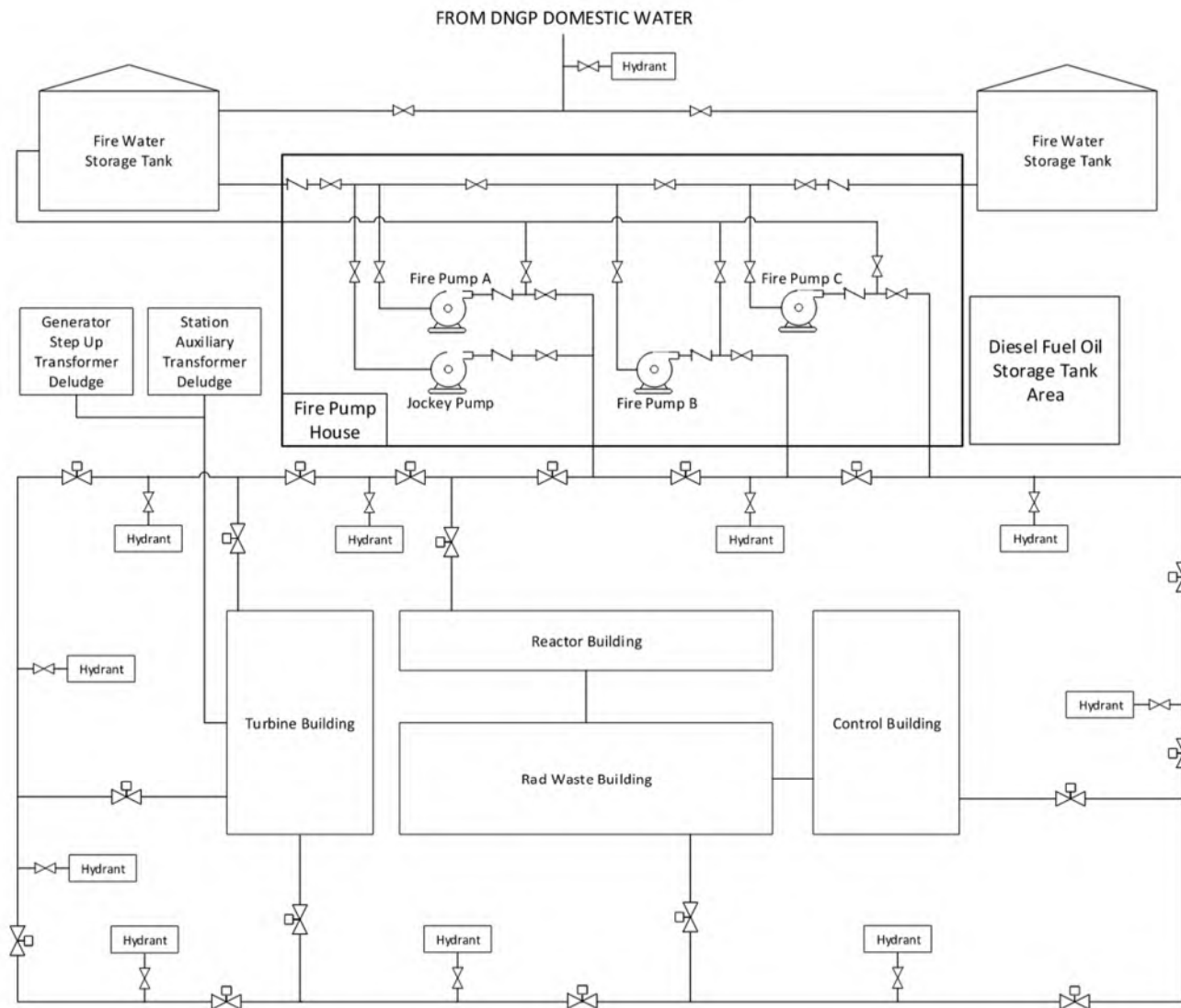
#### **Codes, Standards, and Regulatory Guidance**

Table 9A.6.2-1 lists the codes, standards and guidelines used in the fire protection FPS design.

#### **System Description**

Figure 9A.6.6-1 shows the FPS simplified system diagram for the BWRX-300 Standard Plant facilities. Table 9A.6.2-3 lists the major FPS component design characteristics.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 9A.6.6-1: BWRX-300 Fire Protection System Schematic**

The FPS is the integrated complex of equipment and components that provides early fire detection and suppression to limit the spread of fires. The FPS is part of the overall fire protection program including the plant design and layout to prevent or mitigate fires and includes administrative controls and procedures.

The type of fire suppression is based on the combustible loading and the extent of SC1 or SC2 equipment within a fire area. Use of automatic fire suppression is also determined based on life safety requirements defined in NFPA 101, NFPA 804, CSA N293, CSAN293S1, NBCC or NFCC. Fixed automatic fire suppression systems are installed in areas identified as having a high fire hazard rating by the FHA. Building standpipes and hose stations are provided in major buildings. Portable fire extinguishers are strategically located throughout the plant in accordance with NFPA 10, except in highly radioactive areas.

An automatic fire detection, alarm, supervisory control, and indication system is also provided in selected areas of the plant, as required by the FHA for personnel safety and fire brigade notification.

A Main Fire Alarm Panel (MFAP) located in the Main Control Room (MCR), monitors, and receives system actuation, supervisory, and trouble alarm signals from the individual local panels.

### **Facility Feature for Fire Protection**

Consistent with applicable SC1 or SC2 requirements, structures, systems, and components are designed and located to minimize the probability and effect of fires. To the maximum extent practical, noncombustible, and fire-resistant materials minimize the combustible loading and thereby reduce the expected duration, severity, and intensity of fires.

Exposed structural steel protecting areas containing SC1 or SC2 equipment is fireproofed with material with a fire rating of up to three hours as determined from the FHA.

Access stairwells are enclosed in minimum 2-hour rated firewalls and equipped with self-closing fire-rated doors. Openings in fire barriers or firewalls are equipped with fire doors, frames, and hardware rated the same as the barriers they penetrate.

Seismically supported SC1 or SC2 circuits and circuit routing that contain raceways comply with Branch Technical Position (BTP) SPLB 9.5 1, unless justified under the FHA.

A general intent for BWRX-300 fire protection is to avoid the use of electrical raceway fire barrier systems, relying instead on divisional separation by fire area and structural fire barriers.

### **Fire Protection Water Supply System**

Figure 9A.6.2-1 provides a simplified diagram of the firewater supply piping and supply piping for BWRX-300 Standard Plant facilities.

#### **Water Source**

The FPS includes two (2) firewater storage tanks. These tanks are designed in accordance with NFPA 22. These tanks meet NFPA 804/ CSA N293 requirements for the total water demand for a period of 120 minutes. They have a capacity greater than 2,000,000 liters (526,000 gals.) per NFPA 804/ CSA N293.

Physical separation of redundant storage tanks ensures protection against common cause failures such as seismic events and tank rupture. Each tank is equipped with a freeze protection system as necessary based on local climate conditions.

Fire department connections on all major buildings allow a fire department pumper truck to pump water into the FPS as an additional fire protection water supply source.

Freeze protection is provided for the firewater storage tanks and exposed piping.

### **Fire Pumps**

Three fire pumps each provide 60% of the firewater demand to the worst-case fire within the RB, TB, CB, RWB or Facility Annex Building. Fire pumps are capable of delivering the flow and pressure required to the location that is the most hydraulically remote from the firewater supply source. The fire pumps are located in the yard near the firewater storage tanks. Two of the three fire pumps are required to supply sprinklers and standpipe system. There is a combination of electric and diesel-driven fire pumps. The diesel-driven pump is an air-cooled pump with skid-mounted auxiliaries and a gravity-drain fuel oil supply.

There is a total of three (3) firewater pumps being either electric or diesel. Per CSA N293 (Reference 9A.6.11-1), where more than two pumps are provided, and the largest pump is unavailable for service, then the remaining pumps combined should be capable of providing at least 120% of the necessary, designed flow rate. The designed flow rate is to be based on the scenario of a Turbine fire, which has been determined to be the bounding worst-case fire within the Power Block. Any combination of 2 of the 3 pumps is able to achieve 120% of the needed design flow rate. All fire pumps are capable of delivering the flow and pressure required to the location that has the most limiting combination of distance, line size and/or flow rate. Firewater pumps and accessory components comply with the requirements of NFPA 20. Diesel engines comply with NFPA 20.

An electric fire pump is to be located in the fire-rated firewater pump enclosure (FPE), adjacent to the firewater storage tank and near the power block. The FPE is mounted on a separate foundation. A diesel-driven fire pump is to be located in a fire-rated structure. All pumps are separated by a 3-hour fire-rated fire barrier. Diesel-driven fire pump(s) provide(s) firewater in the event of failure of any electrically driven fire pump or LOOP.

Fire barriers between the fire pumps and their associated equipment as necessary to prevent common cause failures. The diesel-driven fire pump provides firewater in the event of failure of the electrically driven fire pump of the same load group.

The fuel oil day tank for a diesel-driven fire pump is designed in accordance with NFPA 20 and have a capacity sufficient to allow operation of the diesel engine for a minimum of 8 hours before refilling based on fuel consumption at rated pump capacity.

The fuel oil day tank(s) are/is included for the diesel-driven fire pump(s). The fuel oil day tank(s) for the diesel-driven fire pump(s) are/is designed in accordance with NFPA 20 and has a capacity sufficient to allow operation of the diesel engine(s) for a minimum of 8 hours before refilling based on fuel consumption at rated pump capacity. Diesel-driven fire pumps contain chillers that function passively during operation of the pumps.

For a fire within the protected area, an electric pump is designed to start first followed by a second pump, if the first pump fails to start or extra capacity is needed. The third fire pump is designed to start in the event of a failure in either of the other two pumps. Any fire pump may be started manually from the Main Fire Alarm Panel in the control building control room (MCR), from the secondary control room (SCR), or locally. All fire pumps can only be stopped manually.

The associated pressure maintenance (jockey) pump is used to maintain pressure in the system at all times.

Remote signals of fire pump status are provided in the MCR and SCR in accordance with NFPA 20.

The diesel-driven fire pump meets the emissions requirements for the year they are manufactured per NFPA 20.

The non-safety power Distribution System provides power to the motor-driven fire pumps, booster pumps, and jockey pumps. The pumps are powered by the controllers.

### **Firewater Supply Piping, Yard Piping, Valves, Sprinklers, and Yard Hydrants**

Figure 9A.6.6-1 provides a simplified diagram of the firewater supply piping and supply piping for BWRX-300 Standard Plant.

The fire water supply piping consists of a buried yard main loop that feeds supplemental loops in the power block. Check valves are provided between each building connection from the main yard piping loop.

The reactor building piping consists of welded carbon steel piping in accordance with ASME B31.1. The yard main consists of buried Class 200 high-density polyethylene (HDPE) piping or other code compliant pipe material, FM approved for fire main service, in accordance with NFPA 24. Other piping located in buildings consist of carbon steel piping in accordance with NFPA 13, 14 and 15.

The sprinkler systems are designed and installed to meet the requirements of NFPA 13. The deluge systems are designed and installed to meet the requirements of NFPA 15 and/or NFPA 16. Fuel tank foundations and supports meet NFPA 30. Foam injection systems, if installed, are provided to inject foam concentrate into the fire protection water of a foam-water system for distribution over areas protected by foam-water. Foam injection systems are designed in accordance with NFPA 11. Additionally, fixed suppression piping located within the power block is designed and installed in accordance with ASME B31.1. The systems are designed so that an SSE does not cause failures which could impair the functionality of nearby safety class and safe shutdown related structures, systems, and components. Isolation valves which, if closed, would prevent fixed suppression system operation, have tamper switches to monitor their normal position as required by NFPA 13/CSA N293. Isolation valves that control only water supplies to a fire hydrant are post indicating valves and may be locked open in lieu of fire alarm supervision to satisfy CSA N293 (Reference 9A.6.11-1). Fixed suppression systems have provisions to properly drain all parts of the systems. Clean Agent Fire Extinguishing Systems that meet the requirements of NFPA 2001 may be used in addition to the sprinkler-based systems noted below or in areas where sprinkler-based systems are not required per NFPA 804/CSA N293 Fire suppression protection for Canadian BWRX-300 plants meet the provisions given under CSAN293 (Reference 9A.6.11-1).

Standpipes and hose connections in the reactor building are designed to maintain pressure integrity and operability following an SSE in accordance with NFPA 14 and CSA N293. The piping in these areas is designed and installed in accordance with ASME B31.1. Standpipe sizes are at least 150 mm (6 inch) if the building height exceeds 30.5 metres (100 feet). Standpipe sizes for other buildings are at least 100 mm (4 inch) however designed to perform per CSA N293 and NFPA 14 requirements. Drain valves are provided at the bottom of each standpipe to allow for drainage. Fire hoses and accessories are provided in accordance with CSA N293 (Reference 9A.6.11-1) and the FHA. Each hose station has a 65 mm (2-1/2 inch) hose valve with cap rooms and inside entrances for normally occupied rooms.

Fire hydrants located at approximately 75 m (250 ft) intervals along the yard main loop provide firefighting capability, especially in the vicinity of buildings or structures containing combustible materials. The fire hydrants are located no closer than 12.2m (40 ft) from the buildings and structures protected by the hydrants.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

All yard fire main piping is designed and installed to NFPA 24. Fire hydrants are designed and installed to comply with NFPA 24, NFPA 804, and local fire department requirements.

Fire hydrants are protected against freezing and damage from vehicles.

### **Manual Suppression Means**

Manual fire suppression means are provided for all plant areas. The sprinkler systems and the hose station standpipes have separate connections to the firewater main; therefore, no single failure can impair both systems.

### **Standpipe and Hose Systems**

Standpipe and hose stations are provided in all major buildings. Standpipes in areas adjacent to stairways and other locations provide sufficient hose coverage.

The wet standpipes and hose stations are designed to NFPA 14 Class III Service.

Areas containing equipment required for safe shutdown, standpipes and hose connections for manual firefighting remain functional following an SSE. Provisions are made to supply water to at least two standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of an SSE. The piping system serving such hose stations is analyzed for SSE loading and is provided with supports to ensure system pressure integrity. The piping and valves for the portion of the hose standpipe system affected by this functional requirement, as a minimum, satisfy ASME B31.1 Requirements.

All rooms within the plant buildings are within the reach of at least one effective hose stream from a Class III hose station unless evaluated as acceptable under the FHA. Effective hose streams from two separate hose stations cover each room that contains equipment required for safe shutdown that is not protected by a fixed fire suppression system. The need for two-hose-station coverage is also based upon the fire hazard present. Rooms not covered by a fixed fire suppression system and with coverage by only one hose station are furnished with portable fire extinguishers for secondary coverage.

Hose stations also provide secondary coverage for fixed suppression systems.

Hose stations are located outside of highly radioactive areas where possible; however, hose stations are located such that any location that contains or could present a hazard to SC1 or SC2 equipment can be reached by at least one effective hose stream 30.5 metres (100 feet) with a maximum of 30.5 metres (100 feet) of hose.

Standpipes and hose stations external to containment and portable extinguishers provide protection during refueling and maintenance operations. Hose stations are located such that any location within containment can be reached by two effective hose streams with a maximum of 30.5 metres (100 feet) of hose.

Fixed fog hose nozzles protecting high-voltage electrical equipment rooms preclude electrical shock hazards with shutoff capability isolation valves. Adjustable fog and straight stream nozzles are provided for all hose stations located away from high-voltage electrical equipment.

### **Fire Extinguishers**

Portable fire extinguishers for manually extinguishing fires are strategically placed throughout the plant in accordance with NFPA 10 and NBCC, except in highly radioactive areas. Portable fire extinguishers are readily available outside the highly radioactive areas, so that they can be accessible when needed.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Portable Class ABC multipurpose dry chemical-type fire extinguishers are provided for all general hazard areas throughout the buildings in the plant.

Portable Carbon Dioxide Class B, C fire extinguishers are provided for electrical areas.

Special use portable fire extinguishers are provided based upon the hazard present (e.g. – Class D fire extinguishers for a Hot Machine Shop).

### **Water Spray for Charcoal Filters**

Water spray systems are provided for the fire protection of charcoal filters in the HVS, and offgas systems as required. The water is supplied to the filters by means of a fixed piping network extending to a valve station near the exterior of the protected equipment (with manual shutoff valves) and then to open sprinklers or nozzles in the filter enclosure. Drainage is provided for the enclosure.

### **Fixed Automatic Water Extinguishing Systems**

The selection of specific types of fire suppression systems and the areas requiring protection are based on equipment arrangements and combustible loading in each fire area.

Sprinkler piping for areas containing SC1 or SC2 equipment meets the requirements of NFPA 13 (assurance that any failure of FPS piping caused by an earthquake does not damage a SC1 or SC2 item).

Fixed automatic fire suppression systems are provided in all areas identified as having a high fire hazard rating where the combustible loading exceeds 700 MJ/m<sup>2</sup> for non-electrical areas or 1400 MJ/m<sup>2</sup> for electrical equipment areas or elsewhere as required by the FHA.

### **Wet Pipe Sprinkler System**

Automatic sprinklers provide protection for the areas identified in the FHA, except where conditions dictate the use of other types of systems or fire suppression agents.

Each wet pipe sprinkler system consists of an alarm check valve, thermally actuated closed-head sprinklers, and attached piping network containing water under pressure. Water discharges immediately from sprinklers opened by heat from a fire. The wet pipe sprinkler system meets the requirements of NFPA 13.

### **Precision Sprinkler System (Manual or Automatic)**

Precision sprinkler systems are provided for plant areas where it is necessary to prevent serious water damage to equipment or contents that could occur with other sprinkler systems as a result of leaking automatic sprinkler heads, pipe breaks, or inadvertent actuation. Each precision sprinkler system consists of a deluge valve, means to contain precision air (such as a precision check valve), thermally actuated closed-head sprinklers or spray nozzles, and attached piping containing air under pressure (the air pressure serves to monitor the integrity of the piping and provides a supervisory alarm in the event of trouble). Fire detectors are installed in the same room as the sprinklers.

### **Deluge System (Manual and/or Automatic)**

Deluge systems are provided in high fire hazard areas where required by the system design, such as transformers. Deluge systems apply water or foam-water solutions immediately over the entire hazard. Foam-water solutions may be used for areas that contain Class II combustible liquids (such as fuel oil) or flammable liquids (except diesel-driven fire pumps). Each deluge system consists of a deluge valve, open-head sprinklers or nozzles, and a dry piping network which is connected to the firewater supply. Fire detectors are installed to monitor the protected equipment or be in the same room as the nozzles and activate the system.

### **Foam System (As Required)**

A foam system is provided for concentrated fuel oil or lube oil hazards.

Fire detectors are used for fire detection that actuates precision foam-water sprinkler or deluge foam-water spray systems to protect against inadvertent actuation.

A foam hose rack is located outside each diesel generator area for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

A foam hose rack is located outside each diesel-driven fire pumps enclosure area for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

A foam hose rack is located outside the turbine lube oil skid for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

### **Fire Detection and Fire Alarm System**

The fire alarm system is to provide a two-stage operation, as follows: (a) first stage — an alert signal, and (b) second stage — an alarm signal. The alert signal is directed to MCR staff and may remain silent throughout the balance of the building to suit the requirements of the plant's emergency notification procedures. On receipt of an alert signal, MCR staff has the capability to immediately provide a voice announcement over the fire alarm system, throughout the protected



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

area and external areas of the BWRX-300 plant with the exception of the main and secondary control rooms. The alarm signal is activated automatically in the event that MCR staff do not acknowledge the signal within 5 min of initial fire alarm system activation. The alarm signal is supplemented by voice announcements. There is essentially no delay in the ability to override the alarm signal and operate voice communication functions. This fire alarm and voice communication systems with integrated, supervised, one-way voice communications is provided in all structures and exterior areas within the protected area, as well as structures and areas external to the protected area where SSCs directly support the plant. Fire and voice signals are to be distinctive and not easily confused with other alarm signals. Accessible spaces, with the exception of the main and secondary control rooms, are to be equipped with audible and/or visual fire alarm signal devices. Design, installation, and inspection of the fire alarm system comply with CAN/ULC-S524 and CAN/ULC-S537.

Fire detectors for very early warning and annunciation of fire conditions are separate from fire detection and releasing devices for suppression system actuation. The fire detection system is electrically supervised to detect circuit breaks, ground faults, and power failure.

Deluge and preaction fire detection systems have a 90-hour (minimum) backup battery packs (with 10 minutes of alarm) located at the local fire control panel (releasing panels). The remainder of the fire detection and alarm system has a 24-hour (minimum) backup battery packs (with 5 minutes of alarm) located at each local fire control panel (supervisory panels) and at the MFAP. Immediately following this 24-hour period, emergency battery power under full load is available for not less than 2 hours. Primary power is provided from the plant Uninterruptible AC Power Supply.

Manual fire alarm stations (pull box stations) are located at all exits as required by the NBCC. Where the 60-metre (200-feet) exit rule is used, manual fire alarm stations are located along each main aisle so that the maximum travel distance from the aisle to the manual fire alarm station is not more than 30 metres (100 feet) for areas without sprinklers and not more than 45 metres (150 ft) in areas with sprinklers throughout all normally occupied buildings to meet CSA N293 (Reference 9A.6.11-1). Visible fire notification is provided in manned office areas and other areas of the plant where reasonable.

Smoke detection in ventilation systems is provided to meet NFPA 90A/92A requirements. Smoke detection for elevator equipment is provided to meet ASME A17.1/CSA B44 requirements.

Primary Containment is inerted during normal operation. Portable detection equipment and fire watches, as required, are used inside containment during maintenance outages when the space is not inerted.

Manual fire alarm stations (pull box stations) are provided at the normal exit paths or every 61 metres (200 feet), whichever is less, throughout normally occupied buildings. Manual fire alarm stations (pull box stations) are provided at exit paths only for unoccupied buildings. Visible fire notification is provided in manned office areas such as in the CB and the Plant Service Area.

### **Fire Barriers**

Fire barriers of three-hour fire resistance rating are provided separating:

SC1 or SC2 systems from any potential fires in SC3 areas that could affect the ability of SC1 or SC2 systems to perform their safety function.

Redundant divisions or trains of SC1 or SC2 systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

Components within a single SC1 or SC2 electrical division that present a fire hazard to components in another SC1 or SC2 division.

Electrical circuits (SC1, SC2 and SC3) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function.

Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barrier. Only noncombustible materials qualified per CAN/ULC-S102 are used for construction of fire barriers. Fire dampers protect ventilation duct openings in fire barriers as required by CSA N293 (Reference 9A.6.11-1) and CSA N293S1 (Reference 9A.6.11-7).

Design features that prevent or mitigate spurious actuations include:

The FPS is part of the overall plant Fire Protection Program, which ensures the safety of personnel/public, the protection of property/environment, and the continuity of electric power production. In addition to the FPS equipment and components, the Fire Protection Program (FPP) includes combustible inventory control, divisional separation, plant layout and equipment location considerations, fire barriers, training, and administrative controls. The overall Fire Protection Program provides assurance, through defense in depth, that the plant can achieve and maintain a safe shutdown condition in the event of a fire and that radioactive releases to the environment in the event of a fire are minimized.

The Fire Safe Shutdown Analysis (9A.6.11-5) considers multiple hot shorts for safe shutdown equipment that has hard wires for its power or control circuits that could cause equipment to either actuate or not actuate which could result in the equipment not being able to be placed in its safe shutdown position. Hard wires that are in conduit do not have to consider hot shorts from conductors/cables that are outside of the conduit. Hard wires in cabinets/panels that are in wire bundles are considered for hot shorts. CSA N293 / N293S1 separation criteria such as three-hour fire barrier separation between redundant success paths is considered in the safe-shutdown circuit analysis.

The Fire Safe Shutdown Analysis uses a deterministic analysis approach for the safe-shutdown circuit analysis and, therefore, does not use any performance-based approach like fire modeling that considers the location of cables and equipment.

### **Building Ventilation**

Fire protection/smoke control provisions for ventilation for the various building areas are designed as follows:

The Plantwide HVS removes smoke from fire areas. Duct mounted smoke detectors transmit an alarm signal to the local fire control panel upon detection of smoke.

FPS provides fire suppression for the Secondary Control Room (SCR) Makeup AHU filters, annunciation and isolation of Main Control Room (MCR) and Secondary Control Room (SCR) ventilation under fire, smoke purge mode for U41 systems, and interfaces for down-powering ventilation fans from Fire Alarm Panels. The Turbine Building, Reactor Building, Radwaste Building, Control Building and Facility Annex Building are provided with pressurized stairwells to create smoke free egress routes in the event of a fire.

Safe egress and safe smoke refuge areas during a fire incident are provided in accordance with CSA N293 / N293S1 (References 9A.6.11-1 and 9A.6.11-7), NBCC and NFCC guidelines for building occupants and the fire brigade. The Turbine, Reactor, Radwaste Control and Plant Service Area Buildings are provided with pressurized stairwells, in accordance with NFPA 101 and CSA N293 / N293S1 Guidelines are utilized for the design and labeling of safe egress routes.

Smoke removal meets CSA N293 / N293S1 requirements. Automatic sprinkler protection is provided as directed by the FHA for the high-density cable tunnels, fuel oil tank rooms, diesel generator rooms, and a significant portion of the Turbine Building to limit heat and smoke generation.

Procedures for manual smoke control by manual actions of the fire brigade for all plant areas are in accordance with CSA N293 / N293S1 guidelines.

#### **9A.6.7 Instrumentation and Control**

The Fire Protection System (FPS) provides detection, suppression and notification of smoke and fire incidents within the power block and supporting facilities of the BWRX-300.

Each fire suppression system automatically actuated by a fire detection system has the control logic and the capability for manual actuation available at the local fire panel for the protected room. Automatic sprinkler systems which do not require separate detection systems for actuation are generally not equipped with manual actuation means. All instrumentation for fire detection or automatically actuated fire suppression systems is either FM Approved or UL Listed or both.

There are three primary types of FPS instrumentation: instrumentation supporting fire detection, instrumentation supporting automatic suppression systems, and instrumentation supporting firewater delivery.

Instrumentation for the fire detection system provides early detection and warning alarms of fires. Each room containing safety-related equipment has a fire detection system comprised of multiple detectors. Heat and smoke detectors are supervised by local fire panels per NFPA 72/CSA N293 / N293S1 guidelines. The local panels are in turn be connected to the MFAP via a dedicated FPS data link. Signals transmitted include detector status (normal, alarm, supervisory, trouble) as well as local fire panel status. A complete listing of fire detector selection by fire area and/or room number is found in the FHA.

Instrumentation for fixed fire suppression systems provide local and remote monitoring capability for the suppression system status. Alarm check valves are used to indicate that a sprinkler head has opened for wet-pipe sprinkler systems. Pressure/flow sensors are also be used to provide local and remote indication of wet-pipe sprinkler systems. High and low pressure sensors monitor the instrument air pressure in preaction sprinkler systems. High pressure sensors monitor the system pressures downstream of deluge valves. Pressure and level sensors are used for remote and local monitoring of the foam injection systems status. All instruments for automatic suppression systems are wired to the local fire panels for control.

Instrumentation supporting firewater delivery provides status indication of firewater tank level, firewater main pressure, jockey pump status, and main fire pump status conditions.

The MFAP is directly connected to the MCR panels to display common fire alarm, supervisory, and trouble conditions.

The safety class 2 and 3 distributed control and instrumentation system provides HVS system shutdown interfaces from the local fire panels, upon fire detection within HVS ductwork where required by end equipment safety classification.

The non-safety class distributed control and instrumentation system provides HVS system shutdown interfaces from the local fire panels, upon fire detection within HVS ductwork, for equipment not supported by the C20 system.

### **9A.6.8 Monitoring, Inspection, Testing, and Maintenance**

#### **Monitoring**

The FPS is monitored to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

#### **Inspection**

The FPS is periodically inspected to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

#### **Testing and Maintenance**

The FPS is maintained as necessary to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

The following NFPA tests, which are required by code, are performed as a minimum part of the inspection and maintenance requirements for the system:

- a. Waterflow alarm tests of sprinkler systems – NFPA 25
- b. Foam concentration tests, as required – NFPA 25
- c. Deluge waterflow alarm and operational tests – NFPA 25
- d. Foam injection system tests, as required – NFPA 25
- e. Fire detection and alarm system tests – NFPA 72/ CAN/ULC-S537
- f. Smoke removal system/fire damper/smoke dampers – NFPA 92A
- g. Flow testing of fire pumps – NFPA 25

#### **Temporary Configurations**

Temporary configurations controlled by plant procedures.

#### **Required Surveillances**

There are no Technical Specification surveillances required for fire protection.

#### **Maintenance**

The equipment and components of the FPS are designed for easy inspection and maintenance during plant operation. System and equipment manuals are supplied that provide instructions and procedures for installation, operation, and maintenance, and include identification of recommended tools and spare parts.

##### **9A.6.8.1 Post-Maintenance Testing**

Post maintenance testing is controlled by plant procedures.

##### **9A.6.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA) in operational states.

##### **9A.6.10 Performance and Safety Evaluation**

The Fire Protection System does not perform a Safety Category function. The Fire Protection system is sufficiently separated and/or isolated from Safety Category function SSC to preclude

the possibility of a failure in the FPS from impacting the ability of a Safety Category function SSC from performing its Safety Category function. A failure of the FPS cannot initiate an event more severe than has been considered in the safety analyses and cannot degrade the operation of Safety Category function systems.

### **Nuclear Safety Performance**

**Reactor shutdown** - Means are provided to rapidly insert negative reactivity into the reactor core to achieve and maintain subcritical conditions and ensure fuel design limits are not exceeded.

**Decay heat removal** - Means are provided to ensure that fuel is in a safe and stable condition, through the maintenance of sufficient coolant levels and the removal of decay heat from the reactor.

**Barrier to fission product release** – Means are provided to ensure that nuclear reactor systems that contain radioactive materials or fission products, including the reactor coolant system and reactor auxiliary systems, are not breached. There is no leakage of coolant beyond the capability of the pressure and inventory makeup system.

**Support services** – Means for supply the necessary power, water, compressed air, and other support functions is provided to ensure that the above criteria are met.

### **Nuclear Safety Assessment Metrics**

1. Protect the operability of the success path to fire safe shutdown for systems and subsystems, including components and cables, of the fire safe shutdown systems that together are sufficient to meet the nuclear safety performance criteria.
2. Meet the criteria under National Building Code of Canada (NBCC) and National Fire code of Canada (NFCC) for fire barrier, detection, suppression systems and maintain criteria in FHA for areas containing radioactive materials.

### **Life Safety Objectives**

The following life safety performance objectives are met during all operational modes and plant configurations:

1. Fire hazard controls are included in design and operational stages
2. Fire notification means are provided
3. Safe egress and/or areas of refuge are provided to occupants for use in the event of a fire
4. A safe environment and other required supports are provided for essential staff so that they can perform all necessary plant control functions during and following a fire
5. Protection for personnel performing emergency services are provided both during and following a fire
6. Access and emergency lighting is provided for all areas where manual firefighting, evacuations, or operator field actions are expected

### **Life Safety Criteria**

The life safety objectives are met using either the prescriptive requirements or performance-based criteria outlined in the NBCC and NFCC.

### **Life Safety Assessment Metrics**

Life Safety Assessment Metrics are met with the criteria under the prescriptive requirements or performance-based criteria outlined in the NBCC and NFCC.

### **Fire Protection Critical Design Features**

Fire protection systems and practices, and design features to prevent the spread of fires in the BWRX-300, are provided for assurance of safe shutdown capability, prevention of radioactive release to the public, personnel safety, investment protection, and plant availability. These design features include all the following:

1. Noncombustible and heat-resistant materials are used wherever practical throughout the unit.
2. Fire barriers that are rated by approved laboratories in hours of resistance to fire and are used to prevent the spread of fire.
3. SSC important to safety are designed and located to minimize the probability and effect of fires and explosions. The concept of compartmentalization uses passive fire barriers to subdivide the plant into separate areas or to confine the effects of fires to a single compartment or area minimizing the potential for adverse effects from fires on redundant SSC important to safety.
4. The BWRX-300 safety assessment demonstrates that an adequate level of safety is achieved by the design and it can be used to satisfy the CNSC safety objectives. This assessment is also used to determine whether region or province specific design changes are needed. One key safety objective established for the BWRX-300 is the implementation of an explicitly defined D-in-D concept to ensure multiple, independent layers of protection against radiation releases outside the bounds of regulatory limits. As such, the BWRX-300 safety assessment includes an approach to deterministic safety analysis involving layered analyses that correspond to levels of defence.
5. Considering the consequences of a fire in a given fire area during the evaluation of the safe shutdown capabilities of the plant, the BWRX-300 design demonstrates that one success path of SSC that can be used to bring the reactor to hot shutdown or hot-standby conditions remains free of fire damage.
6. The BWRX-300 design layout provides adequate means of access to all plant areas for manual fire suppression. The plant layout allows for safe access and egress to areas for personnel performing safe shutdown operations. Considerations include fire and post-fire habitability in safe shutdown areas, protection or separation from fire conditions of access and egress pathways to safe shutdown SSC, and potential restrictions or delays to safe shutdown area access potentially caused by security locking systems.
7. Systems containing flammable or combustible liquids are designed to minimize leakage of these liquids. In locations where an uncontrolled leakage of the liquid could jeopardize fire safe shutdown systems, the design provides devices to collect, divert, and safely contain leakages from pressurized and non-pressurized components in order to prevent ignition or limit the size of fire and achieve fire safe shutdown.
8. Suitable design of the ventilation systems limit the consequences of a fire by preventing the spread of the products of combustion to other fire areas. Means are provided to ventilate, exhaust, or isolate the fire area as required with consideration

given to the consequences of ventilation system failure caused by the fire, resulting in a loss of control for ventilating, exhausting, or isolating a given fire area.

Refer to Chapter 3, Section 3.5 for additional discussion of Fire Protection Design Features for the BWRX-300 SSCs.

### **Models Used for Design Decisions and Activities**

#### **Deterministic Fire Model**

The BWRX-300 FPA complies with the FPA described in CSA N293, Fire Protection for NPPs [Reference 9A.6.11-1] and CSA N293S1, Supplement No. 1 to N293, Fire Protection for Nuclear Power Plants (application to small modular reactors) [Reference 9A.6.11-7] and takes a deterministic approach that when combustible material is present, a fire is postulated and damage to a Safety Class SSC assessed. As described in CNSC REGDOC-2.4.1, Deterministic Safety Analysis [Reference 9A.6.11-2], deterministic safety analysis methods can be applied to a wide range of plant operating modes and events, including normal operation and abnormal operation resulting from equipment failure, operator errors and challenges arising from events like fires, floods, or earthquakes.

#### **Probabilistic Fire Model**

The BWRX-300 Internal Fires Probabilistic Safety Assessment (IFPSA) is performed using the current information available from the BWRX-300 plant design and procedures, meeting Requirement 4.3 of CNSC REGDOC-2.4.2 [Reference 9A.6.11-4]. Periodic update of the Probabilistic Safety Assessments (PSA) (Requirement 4.4 of REGDOC 2.4.2 [Reference 9A.6.11-4]) will be performed after the plant begins operation.

Probabilistic FP models are also applied to the BWRX-300 design.

These assessments are used in most cases to supplement a deterministic FPA and provide valuable insights into plant design and operation, including the identification of dominant risk contributors, comparison of the options for risk reduction, and consideration of the cost versus risk and benefit analysis.

Fire PSA does not necessarily address compliance with the FP Codes, Standards, and Regulations.

Guidelines for the preparation and regulatory review of PSAs are available in documents such as NFPA 805 and IAEA Safety Report Series No. 10, "Treatment of Internal Fires in Probabilistic Safety Assessment."

#### **Development of Fire Protection Program**

The BWRX-300 FPP is developed and implemented in a coordinated manner that considers the various fire protection activities of different engineering disciplines, functional groups, and other organizations.

The BWRX-300 FPP details how the program is implemented, managed, monitored, and modified during each phase of the life cycle of a plant. Activities specified in the FPP for each phase is discussed in PSAR Chapters 3 and 19.

#### **Methodology for Fire Hazard Assessment**

The BWRX-300 design utilizes the methodology described in CSA N293 [Reference 9A.6.11-1] for the development of the BWRX-300 Fire Hazard Assessment (FHA). This methodology is illustrated in Figure 9A.6.10-1, BWRX-300 Fire Hazard Assessment Methodology.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

The FHA objective is to identify the specific fire hazards and fire protection capabilities in each area of the plant to demonstrate that potential damage is limited by various active and passive fire protection measures, such that the fire protection goals are achieved.

The FHA addresses all plant building areas and outdoor areas with equipment or storage. The FHA address the goals of nuclear safety, life safety, and impact to the environment. The FHA could also address the goal of minimizing economic loss.

There are specific cases of SSC that are exceptions to the application of CSA N293, classified as two types of exceptions. SSC located outside of the PA and directly support the nuclear facility are included in the scope of CSA N293, and SSCs located inside of the PA that have no nuclear safety concerns and do not directly support the nuclear facility are excluded from the scope of CSA N293.

The content of the BWRX-300 FHA includes the following elements and attributes:

The applicability of CSA and CNSC FP requirements and guidance are evaluated.

1. In situ and potential transient fire and explosion hazards, including amounts, types, configurations, and locations of flammable and combustible materials (e.g., electric cable insulation and jacketing material, lube oil, diesel fuel oil, flammable gases, chemicals, building materials and finishes) associated with operations, maintenance, and refueling activities are identified. The continuity of combustible materials (e.g., exposed electrical cables that span the distance between redundant trains), the potential for fire spread, and sources of ignition are identified and described in the analysis.
2. External exposure hazards (e.g., flammable and combustible liquid or gas storage, adjacent industrial facilities or transportation systems, natural vegetation, and adjacent plant support facilities) that could potentially expose SSC important to safety to damage from the effects (e.g., heat, flame, smoke) of fires are identified.
3. The design, installation, operation, testing, and maintenance of automatic fire detection and suppression capabilities is addressed. The FHA describes the level of automatic protection (e.g., water spray density, gaseous agent concentration) provided relative to the specific fire hazards that are identified.
4. The layout and configurations of SSC important to safety are depicted. The protection for safe shutdown systems within a fire area are determined on the basis of the worst-case fire that is likely to occur and the resulting damage. The FHA explains and documents the extent of such damage. The analysis considers the degree of spatial separation between redundant shutdown systems, the presence of in situ and transient combustibles, the available FP systems and features, sources of ignition, and the susceptibility to fire damage of the cables, equipment, systems, and features in the area that are related to safe shutdown.
5. Reliance on and qualifications of fire barriers, including fire test results, the quality of the materials and barrier system, and the quality of the barrier installation is described.
6. Fire area construction (walls, floor, and ceiling materials, including coatings and thicknesses; fireproofing of structural members; area dimensions and volume; normal ventilation and smoke removal capability; and level of congestion as it applies to access for manual firefighting activities) is described. The FHA provides sufficient information to determine that fire areas are properly selected, based on the fire hazards present and the need for separation of SSC important to safety.



NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

7. Manual fire suppression capability, including systems (e.g., hydrants, standpipes, extinguishers), fire brigades, manual firefighting equipment, plans and procedures, training, drills, mutual aid, and accessibility of plant areas for manual firefighting are identified. The FHA lists the location and type of manual firefighting equipment and accessibility for manual firefighting.

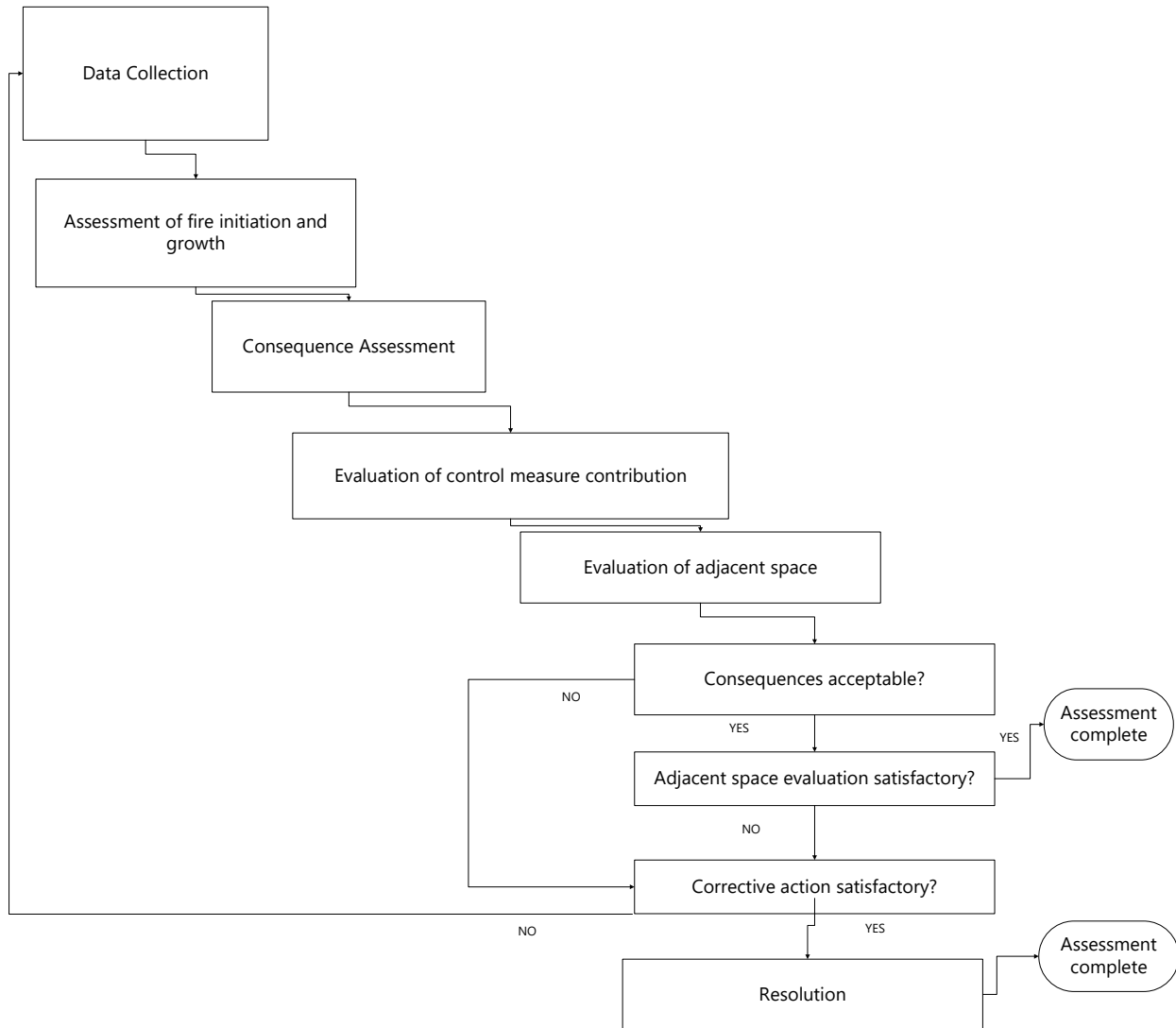
Potential fire impacts on operations are identified, including the following:

1. Fire in control rooms or other locations where operations important to safety are performed
2. Fire conditions that may necessitate evacuation from areas that are required to be attended for safe shutdown, and
3. Lack of adequate access or smoke removal facilities that impede plant operations or fire extinguishment in plant areas important to safety
4. Potential disabling effects of fire suppression systems on safe shutdown capability are identified, including damage to equipment from the normal or inadvertent operation of fire suppression systems. The FHA addresses the effects of firefighting activities.
5. Explosion-prevention measures in areas subject to potentially explosive environments from flammable gases or other potentially energetic sources (e.g., chemical treatment systems, ion exchange columns, high-voltage electrical equipment) are listed.
6. The availability of oxygen (e.g., inerted containment) is identified
7. Alternative or dedicated shutdown capability for those fire areas where adequate separation of redundant safe shutdown systems cannot be achieved is identified.
8. The analysis assumes fire initiation at the location within each fire area or zone that produces the most severe fire with the potential to adversely affect SSC important to safety. Fire development considers the potential for involvement of other combustibles, both fixed and transient, in the fire area. Where automatic suppression systems are installed, the analysis evaluates the effects of the assumed fire, with and without actuation of the automatic suppression system.

The FHA separately identifies hazards and provides appropriate protection in locations where losses of SSC important to safety can occur as a result of the following:

1. Concentrations of combustible contents, including transient fire hazards of combustibles expected to be used in normal operations, such as refueling, maintenance, and modifications
2. Continuity of combustible contents, furnishings, building materials, or combinations thereof in configurations conducive to fire spread
3. Exposure to fire, heat, smoke, or water, including those that may necessitate evacuation from areas that are required to be staffed for safe shutdown
4. Fire in control rooms or other locations having critical functions important to safety
5. Lack of adequate access or smoke removal facilities that impeded plant operations or fire extinguishment in plant areas important to safety
6. Lack of explosion-prevention measures
7. Loss of electric power or control and instrumentation circuits

8. Inadvertent operation of fire suppression systems



**Figure 9A.6.10-1: BWRX-300 Fire Hazard Assessment Methodology**

### **Methodology for Fire Safe Shutdown Analysis**

The BWRX-300 design utilizes the methodology described in CSA N293 [Reference 9A.6.11-1] for the development of the BWRX-300 Fire Safe Shutdown Analysis (FSSA) (9A.6.11-5). This methodology is illustrated in Figure 9A.6.10-2, BWRX-300 Fire Safe Shutdown Analysis Methodology. The FSSA process includes the following steps further defined in NEI 00-01.

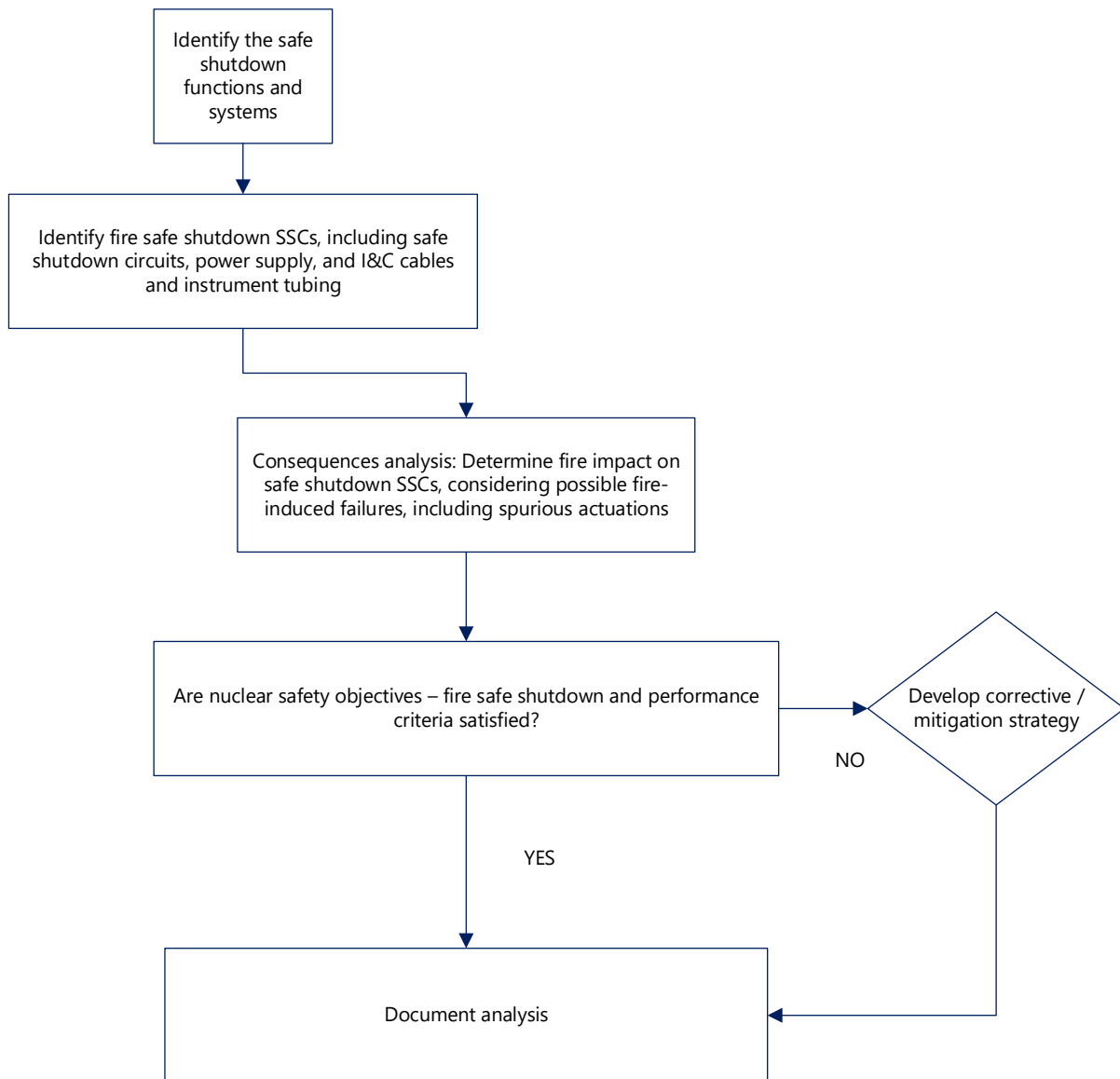
Identify fire safe shutdown systems:

1. Perform consequence assessment to identify the impact of each design basis fire on fire safe shutdown systems and demonstrate that one success path to fire safe shutdown remains available and radioactive materials listed in FHA do not result in unacceptable radiological exposure,
2. Assess manual actions and emergency procedures step required to achieve fire safe shutdown,
3. Consequence acceptability to determine that remaining SSC and cables provide at least one success path to fire safe shutdown
4. Corrective actions if FSSA can not ensure at least one success path remains available for fire safe shutdown including removal or reduction of fire hazards, enhanced fire protection control measures, relocation or protection of exposed FSS component or addition of alternate components

The FSSA demonstrates that at least one means of achieving nuclear safety objectives and performance criteria is available in the event of a fire. The FSSA scope is limited to plant areas where fires have a potential impact on SSC required to perform the fire safe shutdown functions.

It is a general conclusion of this FSSA (Reference 9A.6.11-5) that the station has the capability to safely shutdown the operating unit in the event of a fire. The results of this conceptual study show that the BWRX-300 plant is inherently safe with respect to internal fire events. This is due in large part to the passive safety features of the BWRX-300 plant design which builds on prior ESBWR and Boiling Water Reactor (BWR) Nuclear Power Plant (NPP) design insights.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 9A.6.10-2: BWRX-300 Fire Safe Shutdown Analysis Methodology**

#### **9A.6.10.1 Results of the Third-Party Review of BWRX-300 Fire Protection System**

A third-party review of the BWRX-300 Fire Protection system was conducted by Hatch LTD. (Reference 9A.6.11-9) to ensure that the applicable codes and standards were being applied and that adequate design principles were being utilized as specified by the codes and standards.

Hatch LTD. reviewed the following documents OPG DNNP BWRX-300 Fire Hazard Assessment Requirements Document (Reference 9A.6.11-8), OPG DNNP-1 BWRX-300 Fire Safety Shutdown Requirement and Analysis Document (Reference 9A.6.11-5), and DNNP Preliminary Fire Protection Code Compliance Review Report (Reference 9A.6.11-6). All these documents are currently at the Revision 0 status for Licensing use only.

The Fire Hazards Assessment and Code Compliance Review Report met the requirements described in CSA N293 and CSA N293 Supplement 1 but required more detailed information for final determination of compliance with the Canadian standards. These details will be available as the design of the Fire Protection System and program are completed.

The Fire Safety Shutdown Requirement and Analysis Document complies with Canadian Regulations, Codes and Standards. It follows the methodology in NEI 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis. The documentation for Fire Safe Shutdown is still being developed, but no issue was identified that could impose a significant hurdle to achieving and maintaining fire safe shutdown.

#### **9A.6.11 References**

- 9A.6.11-1 CSA N293," Fire Protection for Nuclear Power Plants, American National Standards Institute," CSA Group.
- 9A.6.11-2 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 9A.6.11-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.6.11-4 CNSC Regulatory Document REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants".
- 9A.6.11-5 NEDC-33978P, "BWRX-300 Darlington New Nuclear Project (DNNP) Fire Safe Shutdown Analysis Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-6 NEDC-33980P, "BWRX-300 Darlington New Nuclear Project (DNNP) Preliminary Fire Protection Code Compliance Review Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-7 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.
- 9A.6.11-8 NEDC-33979P, "BWRX-300 Darlington New Nuclear Project (DNNP) Fire Hazard Assessment Requirements Document," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-9 NEDC-33981P, " BWRX-300 DNNP Independent Third-Party Review Report of Preliminary Fire Protection Design," GE-Hitachi Nuclear Energy Americas, LLC.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.6.2-3: FPS Component Design Characteristics**

Principal Components	Safety Class	Location	Quality Group	Supply Category	Seismic Category	Notes
U43 Fire Protection System	SCN					
Primary Firewater Pump	SCN	FPH	D	TBD	TBD	
Secondary and Tertiary Firewater Pumps	SCN	FPH	D	TBD	TBD	
Firewater Storage Tanks	SCN	YARD	D	TBD	TBD	
Piping, Valves, and Sprinklers	SCN	CB, SCR, seismically qualified access/egress paths, seismically qualified inst. rooms, RB	D	TBD	TBD	
Piping, Valves, and Sprinklers	SCN	Areas containing SSC	D	TBD	TBD	DL 2 for coverage of SSC
Piping, Valves, and Sprinklers	SCN	All other areas	D	TBD	TBD	
Main Fire Alarm Panel	SCN	CB	D	TBD	TBD	
Secondary Fire Alarm Panel	SCN	RB	D	TBD	TBD	

- (1) Fire pump provided in accordance with NFPA 20 with the FP water pumping system design capable of 120% of total required flow at design pressure.
- (2) As a minimum, the fire protection water pumping system consists of at least one diesel-engine-drive fire pump set and one electric-motor-drive fire pump set.
- (3) The water supply system for fire protection is provided with an automatic pressure maintenance method such as jockey pumps independent of the fire pumps.
- (4) The fire protection water supply volume is calculated based on the largest expected flow rate for a period of 2 hr. The expected flow rate is based on the largest concurrent design demand of any automatic water-based suppression system designed in accordance with CSA N293, taking into account that the allowance for manual hose streams (minimum hose stream demand of 2850 L/min (750gpm)).
- (5) The OPG BWRX-300 FPS design meets the requirements of N293 section 7.3.2 for the water source. This includes redundancy. Where reservoirs or tanks are used, two separate reservoirs or tanks, each having 100% of the supply volume required in Clause 7.3.2.1.2, is provided unless a lesser redundancy is deemed acceptable as demonstrated by the FPA. A SMR should require less fire water than a legacy nuclear plant. It is expected that a lesser volume from CSA N293 section A7.3.2.1 recommending 2,000,000 L (526,000 gal) water storage can be justified in the BWRX-300 FHA.



## **9A.7 Supporting Systems for Diesel Generators (Storage and Transfer, Cooling Water and Cooling Air, Starting, Lubrications, Combustion Air Intake and Exhaust)**

### **9A.7.1 System and Equipment Functions**

The BWRX-300 contains two Safety Class 3 (SC3) Standby Diesel Generators (SDGs). These diesel generators provide SC3 electrical power directly supporting DL2 and DL4a functions. The SDGs backup the SC3 busses which directly power SC1 equipment. Ideally, the SDGs power SC1, SC2, and SC3 loads in the same manner (by providing backup power to the SC3 busses). Whether its normal AC power or diesel AC power, it is the same path from the SC3 busses to SC1 equipment, the SC2 equipment, and the SC3 equipment. Further details are provided in Chapter 8.

The auxiliary systems supporting the SDGs include:

- Generator fuel oil storage and transfer system
- Generator cooling water or cooling air system
- Generator starting system
- Generator lubrication system
- Generator combustion air intake system
- Generator emissions

Except for the fuel oil storage and transfer system, these subsystems are provided with the engine skid and do not require auxiliary plant system connections.

The SC3 standby diesel generators are electric start and radiator cooled and do not require plant mechanical support services for operation (such as plant auxiliary cooling or instrument air). Either of the standby diesel generators can support all the SC2 and 3 EDS loads and SC1 EDS loads (three divisions) that require power for completion of Safety Category 1, 2, and 3 functions (i.e., either standby diesel generator can support active decay heat removal). There is onsite fuel storage in large storage tanks and day tanks for each standby diesel generator. The large storage tanks hold enough fuel for at least seven days of operation of the standby diesel generators at required load. The local day tanks are in addition to the seven-day supply of fuel in order to support online testing; the day tanks alone provide up to eight hours of full power operation at rated load.

Standby diesel generator starting time is not important to plant safety. However, when the associated bus loses power, the standby diesel generator automatically starts after a short delay to allow any fast transfer schemes to work. The standby diesel generator start and achievement of rated voltage and frequency is expected (although not credited) to be accomplished within 30 seconds. Specific loads on the A21 and B21 LV busses (and the derivative MCCs) are automatically sequenced after the standby diesel generator successfully starts. The completion of automatic sequencing is expected (but not credited) to take no longer than two minutes. The standby diesel generators can also be manually started/controlled from the MCR, the SCR, and locally.

### **9A.7.2 Safety Design Bases**

For BWRX-300, the SC1 EDS provides at least 72 hours of battery backed AC and DC power, after which it is supported by the standby diesel generators for at least one week, or alternatively, is supported by connection of portable generators. In a complete loss-of-offsite/generator AC

power, batteries are charged via the standby diesel generators or the portable generator connections while also powering the required loads for safety functions.

Although the BWRX-300 does not require any onsite or offsite AC power for safety, the permanently installed standby diesel generators provide standby AC power and generally meet all requirements of the referenced section of the REGDOC. Restoring power as soon as possible is desirable. Following a plant LOOP, the standby diesel generators are expected (although not credited) to automatically start and achieve rated voltage and frequency within 30 seconds. The completion of automatic sequencing is expected (but not credited) to take no longer than two minutes. The permanently installed batteries provide the emergency power required for the SC1, SC2, and SC3 loads to meet plant needs for 72-hours. If offsite power is not restored, AC power is provided via the standby diesel generators.

The SDGs and their auxiliary systems are SC3, who's primary function is to provide power directly to SC3 loads supporting DL2 functions. The SDGs also provides a backup source of power SC1 and SC2 loads.

### **9A.7.3 Description**

#### **Fuel Storage and Transfer System**

A storage tank is provided, which holds sufficient fuel oil to operate the SDGs for a minimum of seven days without refueling. A day storage tank is also provided for fuel oil. Transfer pumps supply fuel oil to each day tank from the fuel oil storage tank. An engine-driven fuel oil booster pump supplies fuel from the day tank to the diesel engine fuel manifold then to the engine fuel injector pumps and injectors.

#### **Generator Cooling Water or Cooling Air System**

Each of the SDGs is equipped with an engine radiator cooling system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required.

#### **Generator Starting System**

Each of the SDGs is equipped with its own dedicated electric start system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required. Dedicated batteries, mounted inside, are Lead Acid batteries.

#### **Generator Lubrication System**

Each of the SDGs is equipped with a dedicated lubrication system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required. All lubricating and fuel handling pumps are API pumps with double seals.

#### **Generator Combustion Air Intake System**

Each of the SDGs is equipped with its own air intake system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required.

#### **Generator Exhaust System**

The Standby Diesel Generator Exhaust system directs diesel exhaust gases away from the SDGs and out of the building. The purpose of this building is to provide protection of the SDGs from ambient environmental effects and to allow the SDGs to perform their functions. The exhaust system design is capable to meet occupational requirements for personnel hearing, emissions, and pollution. Refer to 9A.7.11 on emissions.

#### **9A.7.4 Materials**

Materials for the SDGs and fuel system are purchased in accordance with the requirements of Safety Class 3.

#### **9A.7.5 Interfaces with Other Equipment or Systems**

The SDGs are package skid mounted systems requiring limited interface with other equipment or systems. The SDGs interfaces with plant protective relaying and the DCIS for protection, control, and instrumentation.

The fuel system interfaces with the plant DCIS for control and monitoring.

#### **9A.7.6 System and Equipment Operation**

The SDGs are capable of being manually started and aligned to their respective busses via a local control panel or remotely from the Main and (Secondary) control room. During a loss-of-offsite power, the SDGs automatically start and load on their respective bus via the automatic load sequencer.

Diesel fuel oil is supplied by a delivery system from a 7-day storage tank that is monitored periodically to ensure sufficient fuel oil is available. A supply of lubricating oil is available for the SDGs operating for 7-day operation.

#### **9A.7.7 Instrumentation and Control**

Sufficient instrumentation and controls are available locally to start and operate the SDGs. Additionally, control and instrumentation interfaces are provided to the DCIS such that remote control and monitoring from the Main and (Secondary) Control Room is possible.

Indication and alarms for the fuel supply levels in the fuel storage and day tanks are provided.

#### **9A.7.8 Monitoring, Inspection, Testing, and Maintenance**

Monitoring, Inspection, Testing, and Maintenance activities are provided to ensure that the SC3 SDGs and fuel system can perform their intended support functions.

The SDGs are also redundant and normally in standby. Any required maintenance can be done online. Specific circumstances of standby diesel generator maintenance are dictated by plant technical specifications and controlled by plant procedures.

#### **9A.7.9 Radiological Aspects**

There are no radiological aspects to this system.

#### **9A.7.10 Performance and Safety Evaluation**

The SDGs are sized to carry 100% of the required load following an SBO with sufficient fuel to carry the required load for 7 days.

#### **9A.7.11 Diesel Engine Emissions**

The diesel engine exhaust configuration is covered by NFPA 20 S 11.5.2. Emission data is supplied by the diesel engine manufacturer. Required emission limits are given by local jurisdictions. Diesel engine emissions are rated with the applicable EPA regulations in the year of supply.

#### **9A.7.12 References**

9A.7.12-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

## **9A.8 Overhead Lifting Equipment**

### **9A.8.1 Cranes, Hoists and Elevators**

Cranes, Hoists, and Elevators (CHE) are located throughout the plant and provide the means to lift and lower equipment and materials and move them horizontally along safe load paths. The principal equipment is the Reactor Building Polar Crane, Turbine Building Overhead Bridge Crane and various hoists and monorails. Table 9A.8.1-1 identifies the CHE components.

The Reactor Building Polar Crane is designed to lift heavy loads. A heavy load is defined as a load whose weight is greater than the combined weight of a fuel assembly and the associated handling device. Refer to Chapter 3, Subsection 3.4.4.3 for additional information pertaining to heavy loads.

The CHE provide a safe and effective means for transporting loads including the handling of new and spent fuel, plant equipment, service tools and spent fuel transfer casks and canisters. Safe handling includes design considerations for maintaining occupational radiation exposure as low as practicable during transportation and handling.

#### **9A.8.1.1 System and Equipment Functions**

The system and equipment functions associated with the lifting of loads includes the following:

##### **9A.8.1.1.1 Normal Functions (Non-Safety-Category)**

The CHE system continuously operates during all modes of normal power plant operation, including startup and shutdown, controlling the movement of lifted loads throughout the plant. The CHE system carries out the following functions:

- System for lifting and lowering a load and moving it horizontally, with the hoisting mechanism being an integral part of the system
- Load handling system includes rigging components such as slings, shackles, and eyebolts which connect a load to a lifting device and any lift fixture
  - The CHE system also includes building elevators for passenger and freight movement.

##### **9A.8.1.1.2 Normal Functions (Safety-Category)**

The system does not perform any Safety-Category functions during normal conditions.

##### **9A.8.1.1.3 Off-Normal Functions (Non-Safety-Category)**

The CHE equipment is capable of operation during all modes of off-normal power plant operation.

##### **9A.8.1.1.4 Off-Normal Functions (Safety-Category)**

The system does not perform any Safety-Category functions during off-normal conditions.

The design of the CHE meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 7.15.3 (Reference 9A.8.1-1) as related to lifting and handling of large and heavy loads and ensuring design margin exists as well as interlocks to accommodate lifting of loads. Information pertaining to the impact of hard objects upon SSCs is presented in Chapter 3, Subsection 3.4.4.3.

#### **9A.8.1.2 Safety Design Bases**

The CHE system is classified as SCN. The Reactor Building Polar Crane is designed, fabricated, erected, and tested to appropriate quality standards such that its failure does not impact the function of other Safety-Category function systems.

The Reactor Building Polar Crane is designed to meet the applicable requirements provided in NUREG-0554 (Reference 9A.8.1-2). Structures, Systems, and Components compliant with NUREG-0554 are considered single failure proof. The Reactor Building Polar Crane is designed to support a critical load regardless of the failure of any single component used in normal operation. A critical load as discussed in NUREG-0554 (Reference 9A.8.1-2) is a load being handled by a crane that can be a direct or indirect cause of release of radioactivity.

The consequences of a postulated load drop are considered acceptable when the four evaluation criteria of NUREG-0612 (Reference 9A.8.1-3), Paragraph 5.1, are satisfied.

Plant arrangement and the design of heavy load handling systems are based on the following criteria:

- To the extent practicable, heavy loads are not carried over or near Safety-Category function components, including irradiated fuel and safe shutdown components. Safe load paths are designated for heavy load handling in Safety-Category function areas.
- The likelihood of a load drop is extremely small (that is, the handling system is single failure proof), or the consequences of a postulated load drop are within acceptable limits.
- Single-failure-proof systems can stop and hold a critical load following the credible failure of a single component.
- Single-failure-proof systems can support a critical load during and after a safe shutdown earthquake.

#### **9A.8.1.3 Description**

A description of the CHE system is provided below. Location, safety class and seismic category for system components are summarized in Table 9A.8.1-1.

##### **9A.8.1.3.1 Reactor Building Lifting Devices**

#### **Reactor Building Polar Crane**

The Reactor Building Polar Crane is designed according to NUREG-0554 (Reference 9A.8.1-2) supplemented by ASME NOG-1 (Reference 9A.8.1-4) for a Type I single failure proof crane. Crane Manufacturers Association of America Specification 70 (Reference 9A.8.1-5), and to the applicable ANSI standards.

The Reactor Building Polar Crane is composed of an overhead bridge of two deep girders supporting a trolley with a main and auxiliary hook. The top of a circular rail is located at approximately elevation 22.7 m. This rail supports the two girders allowing the crane bridge 360 degrees of rotation. The bridge structure spans the full width of the refueling floor. The crane is classified as Seismic Interaction to maintain structural integrity and single failure proof to mitigate the probability of a load drop event. Seismic Interaction evaluations ensure that in the event SSCs, fail during a seismic event there are no adverse interactions with the ability of any Seismic Category A or B SSC to accomplish its Safety Class function. Refer to Chapter 3, Subsection 3.2 for additional information pertaining to Seismic Interaction. Design for both the main and auxiliary hoists is in accordance with ASME NOG-1 requirements which provides a high degree of reliability and safety. Redundancy and other features ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold a critical load. The principal heavy loads handled by this crane consist of the components associated with reactor vessel refueling, including the reactor head; reactor vessel internals including dryer/separators; fuel assemblies (associated with new fuel receipt), and various containment support components. During spent fuel handling activities, the cranes maximum load is the combination of the spent

fuel cask, transfer cask, and lifting yoke, which weighs less than the cranes overall capacity. Weight of the spent fuel cask is dependent upon the selected manufacturer. The crane will have access to a truck bay for fuel transfer to spent fuel shipping and/or transfer casks, new fuel assemblies, and replacement components.

#### **Jib Crane**

A jib crane is located in the truck bay near the equipment hatch. The jib crane performs rigging services for replacement of components, materials, and supplies to and from the truck bay to the lower levels of the Reactor Building.

#### **Load Handling**

Overhead pad eyes and rigging beams are located below the concrete slab at elevation 4.9 m. These components are used to manipulate the piping, valves, and equipment at the top of the reactor vessel in the primary containment during outage maintenance activities.

#### **Battery Room Monorails**

A monorail system located above battery racks for battery handling during maintenance activities is provided.

#### **Elevator**

A service elevator is provided.

#### **9A.8.1.3.2 Turbine Building Lifting Devices**

##### **Turbine Building Overhead Bridge Crane**

The Turbine Building Overhead Bridge Crane spans the operating deck above the generator, High Pressure Turbine, Low Pressure Turbines and Condenser. The bridge structure consists of steel girders supporting a trolley with a main hoist and auxiliary hoist. The principal heavy loads handled by this crane consist of components of the turbine-generator system. The heaviest component being the Low Pressure Turbine bladed rotor.

Vertical and horizontal hook travel limits are based on Turbine-Generator equipment requirements as well as the other components located on the operating deck.

CSA B167, (Reference 9A.8.1-6), as well as ASME/ANSI B30.2 (Reference 9A.8.1-7) are used for general construction and installation requirements.

##### **Turbine Building Jib Crane**

The Turbine Building Jib Crane is located at the north end of the operating floor. The Turbine Building Jib Crane performs lifts from ground level to the operating deck for maintenance and operation activities such as transport of inspection materials, toolboxes, and other components.

##### **Diesel Generator Monorails**

Overhead monorails are located above each Diesel Generator. The monorails are used for engine maintenance activities.

##### **Condenser Monorails**

Overhead monorails are located above the circulating water piping elbows directly east of the condenser.

### **Battery Room Monorails**

A monorail system located above battery racks for battery handling during maintenance activities is provided.

### **Hot Machine Shop Overhead Bridge Crane**

An overhead bridge crane is provided in the hot machine shop for maintenance support services.

### **Elevator**

A service elevator is provided.

#### ***9A.8.1.3.3 Radwaste Building Lifting Devices***

### **Tank Filter Monorail**

A Tank Filter Monorail hoist is located above concrete hatches adjacent to the concrete tank wall on elevation 13m of the Radwaste Building. The Tank Filter Monorail hoist is used to lift concrete hatches above the condensate pre-filter tanks.

### **Elevator**

A service elevator is provided.

#### ***9A.8.1.3.4 Control Building Lifting Devices***

Battery room monorails in the Control Building are provided for battery handling during maintenance activities.

#### ***9A.8.1.3.5 Power Block Pre-Qualified Lift Points***

Existing steel or concrete structures throughout the plant have pre-qualified lift points for maintenance activities. The rated load for each lift point is appropriately tagged or stenciled to the building concrete or steel member. Rigging attached to lift points allow for equipment to be moved for repair and/or replacement maintenance activities.

#### ***9A.8.1.3.6 Component Description***

This subsection describes CHE components.

### **Hook**

The hook is a device for grabbing and lifting loads by means of a device such as a hoist or crane. A lifting hook is typically equipped with a safety latch to prevent the disengagement of the lifting wire rope sling, chain or rope to which the load is attached.

The hook is designed to withstand stress imposed under normal operating conditions while handling loads within the rated load. The minimum hook design factor conforms to those specified for the equipment or system in which the hook is a component.

### **Monorails**

Monorails are used for lifting and lowering a load and moving the load horizontally, suspended from a single track. The combined load on all hoists on the monorail do not exceed the rated load of the monorail.

Monorail systems conform to the minimum design parameters as specified in The Steel Construction Manual (Reference 9.8.1-8), Crane Manufacturers Association of America No. 74 (Reference 9A.8.1-9), ANSI MH27.1 (Reference 9A.8.1-10), or ANSI MH27.2 (Reference 9A.8.1-11), as applicable.

## **Hoist**

A hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load. The hoist and appurtenances are designed to withstand the stresses imposed under normal operating conditions while handling loads within the rated load.

### **Hand Chain-Operated Chain Hoist**

A hand chain-operated chain hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load.

### **Electric Powered Chain Hoist**

An electric powered chain hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load.

### **Below-the-Hook Lifting Devices**

A below-the-hook lifting device is used for attaching loads to a hoist. The device contains components such as slings, hooks, and rigging hardware. The design of below-the-hook-lifting devices are in accordance with ASME BTH-1 (Reference 9A.8.1-12).

#### **9A.8.1.4 Materials**

Cranes are fabricated using materials that are designed to meet operating stress limits plus margin. The CHE System equipment and associated structural components are designed to meet the 60-year plant life, with appropriate provisions for maintenance and replacement.

#### **9A.8.1.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.8.1-2 for Cranes, Hoists, and Elevators interfaces with other equipment or systems.

#### **9A.8.1.6 System and Equipment Operation**

##### **9A.8.1.6.1 Normal Operations**

Operators are trained, qualified, and cognizant of the load handling equipment interlocks and protective devices. Interlocks and protective devices are not overridden or bypassed unless authorized by an approved Work Order. Elevators that serve floor elevations above the first floor that are equipped with an automatic emergency recall feature are equipped with smoke detectors installed in the elevator lobbies on the recall level. In-car emergency switches are provided for in all elevator cars.

##### **9A.8.1.6.2 Off-Normal Operations**

Load handling systems are operational unless scheduled for a maintenance.

#### **9A.8.1.7 Instrumentation and Control**

##### **9A.8.1.7.1 Instrumentation**

The following information does not pertain to manual or portable hoists in the CHE system.

The Reactor Building Polar Crane and Turbine Building Overhead Crane control circuits are arranged so that tripping of an overload relay or limit switch defeats the permissive to all crane controllers. Overload relay or limit switch operation does not interrupt lighting circuits.

The avoidance of two-blocking for the RB Polar Crane is accomplished using single-failure-proof features and does not rely on any action by the operator. The normal hoist limit switch is



supplemented by an independent final hoist limit switch operated by the load block to remove power from the hoist motor and brakes.

The prescribed path for Reactor Building Polar Crane hook travel is enforced by limit switches for hook travel in both vertical and horizontal directions.

#### **9A.8.1.7.2 Control**

The following information does not pertain to manual or portable hoists in the CHE system.

The Reactor Building Polar Crane and Turbine Building Overhead Cranes are controlled by a local master control panel on the crane bridge. Primary control of crane motions is directed from a cab located on the crane bridge and remote control with backup pendant control. The cab, remote and pendant controls provide for bridge, trolley, main hoist, hook rotation (cab and radio control only), and auxiliary hoist motions. Pendant controls are back up for cab/remote and are isolated during normal operation by a transfer switch on the pendant. Remote control utilizes lever switch technology.

The Reactor Building Polar Crane electrical system is designed so it is possible for the operator to stop and hold a critical load regardless of the failure of any single component used in normal operation. The operator can stop all motors without a time delay using the emergency stop.

The Reactor Building Polar Crane control system uses a programmable logic controller, housed in a bridge control panel. The programmable logic controller is integral to a restricted zone scheme for the crane which is activated during critical lift evolutions such as movement of the spent fuel cask.

#### **9A.8.1.8 Monitoring, Inspection, Testing, and Maintenance**

Elevators are registered in compliance with Technical Standards and Safety Authority.

Surveillances of a load handling system are performed in accordance with plant load handling procedures. The surveillance is a visual check of the overall configuration of the load handling system. Inspections and tests of the load handling system is performed in accordance with Operations and Maintenance Manual requirements as well as load handling procedures.

A maintenance program based on manufacturers' recommendations, integrating proactive, reactive, preventive, and predictive maintenance and operating experience, is established to increase the probability that CHE structures, systems and components function in the required manner over the design life cycle. The program includes procedures which ensure that records are retained, and test and inspection discrepancies are documented and corrected. Any crane, hoist or monorail found in an unsafe operating condition is tagged out and removed from service until repaired. All repairs are made by qualified personnel in accordance with the manufacturers' instructions. Rigging hardware used for load handling systems is maintained in a similar fashion.

Maintenance activities for CHE load handling systems which are either preventive or the result of adverse conditions determined through an inspection program are corrected by adjustment, repair, or replacement of components before continuing the use of the system/component. Operational testing is performed in accordance with plant procedures.

CHE load handling systems which are altered, repaired, and modified are required to be operationally tested for the functions affected by the alteration, repair, or modification, as determined by a qualified person. In addition, load testing of altered, repaired, and modified load handling systems, may be limited to the functions affected by the alteration, repair, or modification, as determined by a qualified person.

#### **9A.8.1.9 Radiological Aspects**

For purposes of meeting ALARA guidelines the Cranes, Hoists and Elevators include hoisting/transport mechanisms to handle parts and components in decontamination areas, in the contaminated storage areas and in the active workshops.

The Cranes, Hoists and Elevators use remotely operated electric hoists where possible for tasks/equipment associated with general area dose rates in excess of 1 mSv/hr.

Refer to Chapter 12, Subsection 12.1.5.4 for information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.8.1.10 Performance and Safety Evaluation**

The Reactor Building Polar Crane is designed to be single failure proof to improve operational reliability, which is generally implemented by adding design margin, and by increasing material inspection and testing above that found in Crane Manufacturers Association of America Specification 70 (Reference 9A.8.1-5) and other standards invoked by Crane Manufacturers Association of America Specification 70. The RB Polar Crane electrical system is designed so that it is possible for the operator to stop and hold a critical load regardless of the failure of any single component used in normal operation.

Based upon guidance provided in NUREG-0612 (Reference 9A.8.1-3); Safe load paths are defined. Use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance of the crane are provided to ensure safe crane operations.

#### **9A.8.1.11 References**

- 9A.8.1-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.8.1-2 USNRC NUREG-0554 "Single-Failure-Proof Cranes for Nuclear Power Plants."
- 9A.8.1-3 USNRC NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
- 9A.8.1-4 ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.
- 9A.8.1-5 Crane Manufacturers Association of America, Specification 70 "Specification for Top Running Bridge and Gantry-Type Multiple Girder Electric Overhead Traveling Cranes."
- 9A.8.1-6 CSA B167, "Overhead Cranes, Gantry Cranes, Monorails, Hoists and Jib Cranes," CSA Group.
- 9A.8.1-7 ASME/ANSI B30.2, "Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist," American Society of Mechanical Engineers.
- 9A.8.1-8 American Institute of Steel Construction, "Steel Construction Manual."
- 9A.8.1-9 Crane Manufacturers Association of America, Specification 74, "Specification for Top Running and Under Running Single Girder Electric Traveling Cranes Utilizing Under Running Trolley Hoist."
- 9A.8.1-10 ANSI MH27.1, "Specifications for Patented Track Underhung Cranes and Monorail Systems," American National Standards Institute.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

- 9A.8.1-11 ANSI MH27.2, "Specifications for Enclosed Track Underhung Cranes and Monorail Systems," American National Standards Institute.
- 9A.8.1-12 ASME BTH-1 "Design of Below-the-Hook Lifting Devices," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.8.1-1: Cranes, Hoists, and Elevators Components**

<b>Principal Components</b>	<b>Safety Class</b>	<b>Building</b>	<b>Seismic Category <sup>(1)</sup></b>
RB Polar Crane – Main Hook	SCN	RB	Interaction
RB Polar Crane – Auxiliary Hook	SCN	RB	Interaction
Jib Crane	SCN	RB	Interaction
Monorail - Battery Rooms	SCN	RB	Interaction
Elevator	SCN	RB	Interaction
Overhead Bridge Crane Main Hook	SCN	TB	Non-Seismic
Overhead Bridge Crane Auxiliary Hook	SCN	TB	Non-Seismic
Diesel Generator Monorails	SCN	TB	Non-Seismic
Jib Crane	SCN	TB	Non-Seismic
Monorail - Battery Room	SCN	TB	Non-Seismic
Overhead Bridge Crane Hot Machine Shop	SCN	TB	Non-Seismic
Elevator	SCN	TB	Non-Seismic
Monorail - Condenser	SCN	TB	Non-Seismic
Monorail – Filters	SCN	RWB	Non-Seismic
Elevator	SCN	RWB	Non-Seismic
Monorail - Battery Room	SCN	CB	Non-Seismic

(1) Refer to Chapter 3, Subsection 3.2.3 for information pertaining to seismic classification.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.8.1-2: Cranes, Hoists and Elevators System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides power	Crane bridge or Elevator control panel
	The Turbine Building Overhead crane and the Reactor Building Polar Crane are supplied with an A or B train electrical power transfer switch	Crane bridge
	Communications to elevators	Elevator car enclosure
Reactor Building Structure	The Reactor Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for crane or monorail, Elevator supporting steel or concrete structure
Turbine Building Structure	The Turbine Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for crane or monorail, Elevator supporting steel or concrete structure
Control Building Structure	The Control Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for monorail
Radwaste Building Structure	The Radwaste Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for monorail, Elevator supporting steel or concrete structure
Fire Protection System	The Fire Protection System provides for elevator emergency functions, i.e., smoke detectors installed in elevator lobbies for auto return	Elevator car enclosure

#### **9A.8.2 Fuel Building Crane**

The BWRX-300 does not have a Fuel Building and as such does not have a Fuel Building Crane. New fuel is received onsite and is brought into the Reactor Building (RB) through RB hatch. Spent fuel is removed from the Fuel Pool in a cask. Cask loading and transport activities involve use of the Reactor Building Polar Crane located inside of the Reactor Building. Refer to Subsections 9A.1.1 "New Fuel Storage and Handling System," 9A.1.2 "Fuel Storage and Handling System," and 9A.1.4 "Handling Systems for Fuel Cask Loading," and 9A.8.1 "Cranes, Hoists and Elevators," for additional information pertaining to new and spent fuel handling operations.

## **9A.9 Miscellaneous Auxiliary Systems**

### **9A.9.1 Communication Systems**

The communications systems provide effective intraplant communications and effective plant-to-offsite communications during normal operation, maintenance, transients, fire, and accidents conditions including LOOP.

A typical Communication System consists of the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private branch exchange system
- Sound-powered system
- Emergency offsite communications
- Security communication system

#### **9A.9.1.1 System and Equipment Functions**

System and equipment functions associated with the Communication System include the following:

##### **Normal Functions (Non-Safety-Category)**

The communications systems provide intraplant and plant-to-offsite communications during normal operation.

##### **Normal Functions (Safety-Category)**

The communications systems provide no Normal Safety-Category functions.

##### **Off-Normal Functions (Non-Safety-Category)**

The communications systems provide intraplant and plant-to-offsite communications during transients, fire, accidents, off-normal phenomena, and security-related events.

##### **Off-Normal Functions (Safety-Category)**

The communications systems provide no Off-Normal Safety-Category functions.

The Communications System is designed considering the requirements of REGDOC-2.5.2 (Reference 9A.9.1-1) Section 7.20 "Escape routes and means of communication" by providing diverse methods of communication within the plant and in the immediate vicinity as well as to offsite agencies. In addition, the Communications Systems are designed to provide secure communications channels to emergency support facilities and offsite response organizations consistent with the requirements of REGDOC-2.5.2 Section 8.10 "Control Facilities."

#### **9A.9.1.2 Safety Design Bases**

The Communication System does not perform or ensure any Safety-Category function, and thus has no safety design bases.

#### **9A.9.1.3 Description**

The communication system provides the means to conveniently and effectively communicate between various plant locations and with offsite locations during normal, maintenance, transient, fire, and accident conditions under maximum potential noise levels. The communication system

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

allows guards and watchmen on duty to maintain continuous communication with personnel in manned alarm stations, and offsite/onsite agencies. This is typically accomplished by either private automatic branch (telephone) exchange or wireless communication system.

The Communication System is designed to support Emergency Preparedness and Response requirements for accident conditions, including notification of personnel and implementation of evacuation procedures. This capability includes communications support to both onsite and offsite emergency centers as well as provisions for onsite and offsite communications system, each with a backup power source. Refer to Chapter 19 for information related to Emergency Preparedness and Response.

The Communication System is designed to support the coordination between plant personnel during accident or incident conditions under maximum potential noise levels. This capability also includes communications support for firefighting, including support of alternative and dedicated shutdown capabilities. Refer to Subsection 9A.6 for information related to Fire Protection.

The Communication Systems are designed to support coordination of security activities both within the plant and with external security and law enforcement organizations.

The Communications Systems are designed to generally recognized codes and standards.

The Communication Systems are separated from one another, such that a failure in one system does not degrade the performance of the other systems.

The Communication System is designed to remain functional during a LOOP.

The Communications Systems use conventional equipment that have a history of successful operation at similar industrial settings.

On-site plant communications systems are backed up by the standby diesel generators during a LOOP. Most of the communications equipment is located in the Control Building and the Communications Room both of which have standby diesel generator backed redundant power panels and Uninterruptible Power Supply (UPS) backed redundant power panels. Portions of the Communication Systems are operated on battery backed 72-hour uninterruptible power supplies.

#### **9A.9.1.3.1 Component Description**

##### **Wireless Telephone System**

A typical wireless telephone system consists of wireless portable handsets, hands-free type portable headsets, a comprehensive antenna system, and necessary electronics equipment. The wireless telephone system is the primary means of communication for plant operations and maintenance personnel. The telephone-page, private branch exchange telephone, and sound-powered communication systems are for general plant communications and serve as a backup to the wireless system.

A typical wireless telephone system has the ability to dial fixed private branch exchange telephone stations and vice versa. The wireless system has the capability to access the telephone-page system and the capability to access offsite emergency communication links.

##### **Telephone/Page System**

A typical telephone/page system consists of handsets, amplifiers, loudspeakers, and associated electronic equipment. The system supports paging and party lines without crosstalk or interference.



### **Private Branch Exchange System**

A typical private branch exchange system provides communication between the system stations, with capability for transferring calls and providing conference calls. The private branch exchange system typically interfaces with the following communication systems:

- The wireless telephone system
- Redundant connectivity to commercial telephone services
- Access to the telephone page system
- Direct extensions from the private branch exchange locations exterior to the plant

The separate, redundant connections between the private branch exchange and commercial telephone service provides communications between the Main Control Room and the headquarters or other facilities in case of a single fault.

Commercial telephone lines are provided by the local area telephone company.

### **Sound-Powered System**

The sound-powered telephone system provides voice service to key locations throughout the plant. This system uses portable sound-power telephones that can plug into local terminal jacks. The system allows uninterrupted communications. The system does not rely on external power supply for operation.

### **Security Communication**

A typical security communication system utilizes the common site private branch exchange, public address, and sound power phone circuits to facilitate a portion of their communications needs. Typically, Security has a dedicated radio system providing continuous communications between each onsite security officer, watchman, or armed response individual and an individual in each continuously manned alarm station. These radio systems are typically powered by the security power system.

### **Emergency Communications**

There is an emergency communications workspace provided in the MCR to accommodate two people. The emergency communications workspace contains space for communications equipment and administrative tasks.

#### **9A.9.1.4 Materials**

The Communication System is designed using materials that ensure operability in nuclear and industrial environments.

#### **9A.9.1.5 Interfaces with Other Equipment or Systems**

The communication system derives its power from the Electrical Distribution System. Refer to Chapter 8, Subsection 8.1 for information pertaining to the Electrical Distribution System.

#### **9A.9.1.6 System and Equipment Operation**

##### **Normal Operation**

The Communication Systems operation is checked as part of normal daily usage.

Voice and data communications systems and equipment are provided to support all phases of plant operations and maintenance, including emergency operations.

Portable, wireless communication capability is provided using equipment typically supported by base stations, antennas, amplifiers, and repeaters. This is typically the means of communication among control room operators, equipment operators, radiation protection, and maintenance technicians for routine and emergency operations, including surveillance tests, startup and shutdown operations, refueling, job coverage, and emergency or accident conditions.

Communication channels accommodate the expected message load including task requirements in critical or emergency situations, plus allow margin for expansion and contingency. The system provides the capability for open channel, or "party line" communication.

The Communications Systems function in all ambient noise level environments. The equipment allows communication from high-noise areas consistent with performing other tasks in those areas. Adequate means are provided to alert personnel to the use of communication equipment.

The design of the Communication Systems takes into account the applicable requirements of NUREG-0700 (Reference 9A.9.1-2).

#### **Off-Normal Operation**

The Communication Systems will fail in a predictable manner with failure alarms alerting users to failed or degraded system status. The system is checked for functionality as part of daily usage; therefore, system degradation is self-revealing.

##### **9A.9.1.7 Instrumentation and Control**

No special instrumentation is required for the Communication System.

##### **9A.9.1.8 Monitoring, Inspection, Testing, and Maintenance**

Communication Systems equipment is designed to operate reliably within the environment in which it is installed including environmental conditions such as temperature, humidity, radiation, and noise. Furthermore, the Communication Systems are designed to operate taking into account placement of barriers such as shield walls. Communication System equipment is accessible to personnel for operation, inspection, maintenance, and testing.

The Communications System is pre-operational tested. The systems described above are conventional and have a history of successful operation at similar plants. These systems are used and maintained routinely to ensure their availability.

The power sources for the plant page/party-line telephone system and the private branch exchange are tested separately during the pre-operational and startup test program. Measurements or tests required to identify long-term deterioration are performed on a periodic basis.

##### **9A.9.1.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

##### **9A.9.1.10 Performance and Safety Evaluation**

The Communication Systems do not perform Safety-Category functions. The failure of any Communication System does not adversely affect safe shutdown capability. It is not necessary for plant personnel in Safety-Category function areas of the plant to communicate with the control room to achieve safe shutdown of the plant.

Diverse Non-Safety-Category function power supplies connected to the plant standby diesel generators, power the plant page/party-line telephone, Private automatic branch (telephone) exchange and plant radio systems. Failure of any or all of its components does not affect any Safety-Category functions.

#### **9A.9.1.11 References**

9A.9.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

9A.9.2-1 NUREG-0700, "Human-System Interface Design Review Guidelines"

#### **9A.9.2 Lighting and Servicing Power System**

The plant Lighting and Servicing Power System includes normal and emergency lighting in addition to providing service power throughout the plant. The normal lighting provides illumination during plant operating, maintenance, and test conditions. The emergency lighting provides illumination in areas where emergency operations are performed upon loss of normal lighting. Security aspects of the Lighting and Servicing Power System are considered Security Related Information and are addressed in the Security Annex.

##### **9A.9.2.1 System and Equipment Functions**

###### **Normal Functions (Non-Safety Category)**

The Lighting and Servicing Power System provides illumination throughout the plant as required during Non-Safety Category normal conditions.

###### **Normal Functions (Safety Category)**

The Lighting and Servicing Power System does not perform a Safety Category function during normal conditions.

###### **Off-Normal Functions (Non-Safety Category)**

The Lighting and Servicing Power System provides illumination during Non-Safety Category, off-normal conditions, including fire, transient and accident conditions.

###### **Off-Normal Functions (Safety Category)**

The Lighting and Servicing Power System does not perform a Safety Category function during off-normal conditions.

The design of the Lighting and Servicing Power System is consistent with the requirements of REGDOC 2.5.2, (Reference 9A.9.2-1) Section 7.12.2 relative to lighting infrastructure, Section 7.20 relative to emergency lighting, and Section 8.10 relative to Control Facility lighting.

##### **9A.9.2.2 Safety Design Bases**

The Lighting and Servicing Power System does not perform or ensure any Safety Category function, and thus has no safety design bases.

The Lighting and Servicing Power System does not perform any Safety Category functions during or after a design basis accident. Failure of the lighting systems does not compromise Safety Category functions from being performed, nor does it prevent the safe shutdown of the reactor.

The plant illumination levels provided by the lighting systems are in accordance with the applicable lighting levels specified in NUREG-0700 (Reference 9A.9.2-2).

### 9A.9.2.3 Description

The lighting equipment is designed to provide illumination throughout the plant that is equal to or greater than those recommended by the Illuminating Engineering Society of North America.

Emergency lighting is provided in areas where emergency operations are performed and for personnel safety during a power failure. The individual rooms receive lighting appropriate to their contents. The Lighting and Servicing Power System is comprised of the following lighting systems:

- Normal lighting system - The normal lighting system is used to provide illumination in all areas of the plant except the Main Control Room and the Secondary Control Room (the emergency lighting system provides both normal and off-normal lighting in the control room). The normal lighting system is available under normal plant operating, maintenance, and testing conditions and during off-normal conditions if normal AC power remains available. The normal lighting system does not have a backup power source.
- Emergency lighting system - The emergency lighting system is used to provide acceptable levels of illumination in the control rooms during normal operating conditions while also providing acceptable levels of illumination in the control rooms and other areas of the plant when the normal lighting system is not available. Emergency lighting is available in areas where emergency operations are performed, including the access and egress routes to and from those areas. Upon loss of the normal lighting system (including loss of power events), the emergency lighting system provides illumination for the control rooms, remote shutdown areas, battery rooms, and containment. Emergency lighting has a backup source of power that is either diesel backed or battery backed.

The BWRX-300 uses Light Emitting Diode (LED) lighting that is operated from low voltage power panels where the lighting loads are shared with other loads (e.g., controllers and convenience outlets). These various power panels are powered by three sources:

- Offsite power (derived from busses A1 and B1 – Non-Safety Electrical Distribution System)
- Standby diesel generator (derived from busses A21 and B21 – Safety Class 2 and 3 Electrical Distribution System)
- Uninterruptible Power Supplies – either Safety Class 2 and 3 Electrical Distribution System or Safety Class 1 Electrical Distribution System that are backed by 72-hour batteries and most closely associated with emergency lighting

These panels are distributed in most rooms such that lighting for that room can be easily obtained. Extensive emergency lighting is provided since LED loads are small and insignificant to the diesel or battery loading.

The lighting is designed in accordance with applicable industry standards and is provided in all areas using an appropriate mixture of SCN, SC3 diesel backed and SC2 and SC1 battery backed lighting for the specific plant areas.

The lighting equipment and installation inside and outside the control room is designed to remain functional during design basis events and withstand the seismic loads of a design basis earthquake. Lighting fixtures located in the vicinity of Safety-Category function equipment is supported so that they do not adversely impact the equipment when subjected to seismic loading of a safe shutdown earthquake.

The MCR and SCR lighting over the SC1 video display units and the communications console are powered from the Safety Class 1 Electrical Distribution System. The remaining MCR and SCR lighting is split between the redundant diesel backed Safety Class 2 and 3 Electrical Distribution System power feeds. All control room lighting is part of the emergency lighting system since all control room lighting has a backup power source.

#### **9A.9.2.4 Materials**

The plant lighting is designed using materials that ensure operability in nuclear and industrial environments.

#### **9A.9.2.5 Interfaces with Other Equipment or Systems**

The plant lighting derives its power from the Electrical Distribution System (Chapter 8, Subsection 8.1).

#### **9A.9.2.6 System and Equipment Operation**

The plant lighting includes normal and emergency lighting. The normal lighting provides illumination during all plant operating conditions, including off-normal conditions if normal AC power is available. The emergency lighting provides illumination in the control rooms during all normal and off-normal conditions while also providing illumination in other areas of the plant during off-normal plant operating conditions when the normal lighting system is not available. Off-normal plant conditions for emergency lighting include fire, transient and accident conditions.

#### **9A.9.2.7 Instrumentation and Control**

No special instrumentation is required for the lighting system. The emergency lighting automatically actuates on the loss of normal plant lighting.

#### **9A.9.2.8 Monitoring, Inspection, Testing, and Maintenance**

Pre-operational testing verifies that the normal lighting system provides illumination under normal plant operating, maintenance, and testing conditions, and that the emergency lighting system provides illumination where required throughout the station, including areas where emergency operations are performed. System operability is demonstrated by normal use during plant operation. The normal lighting system and the control room portion of the emergency lighting system is normally energized continuously and requires only routine maintenance or testing.

#### **9A.9.2.9 Radiological Aspects**

There are no radiological aspects associated with the design and operation of the lighting system.

#### **9A.9.2.10 Performance and Safety Evaluation**

The lighting equipment used is conventional equipment that has a history of successful operation at similar nuclear and industrial settings. This equipment is used and maintained routinely to ensure its availability.

The lighting system has no Safety-Category functions and therefore requires no nuclear safety evaluation. Failure of the lighting system or its components will not affect the ability of Safety Category systems to perform their intended function.

#### **9A.9.2.11 References**

- 9A.9.2-1 CNAS Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.2-2 USNRC NUREG-0700, "Human-System Interface Design Review Guidelines."

### **9A.9.3 Floor Drain System**

The purpose of the Equipment and Floor Drain System is to provide equipment to drain, collect, and transport liquid waste from the floor drains to the Liquid Waste Management System (Chapter 11, Subsection 11.2) for processing.

#### **9A.9.3.1 System and Equipment Functions**

##### **9A.9.3.1.1 Normal Functions (Non-Safety-Category)**

The EFS is designed to perform the following:

Collect radioactive or potentially radioactive liquid wastes generated in the plant during normal operation, anticipated operational occurrences, startup, hot shutdown, cold shutdown, and refueling, and transfer these wastes to the Liquid Waste Management System for processing or to an appropriate disposal system.

Collect oily wastes from the Electro-Hydraulic Control Unit, Diesel Generators, and Turbine Lube Oil skid during all modes of plant operation for transfer to the SWM (Chapter 11, Subsection 11.4).

##### **9A.9.3.1.2 Normal Functions (Safety-Category)**

The system does not perform any Safety-Category functions during normal conditions.

##### **9A.9.3.1.3 Off-Normal Functions (Non-Safety-Category)**

The system does not perform any Non-Safety-Category functions during off-normal conditions. The system is designed to collect spills and leaks via floor drains during off-normal conditions.

##### **9A.9.3.1.4 Off-Normal Functions (Safety-Category)**

As part of Defense Line 3 the EFS provides containment isolation valves on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Subsection 7.3.1). The EFS can perform this function during and after a Design Basis Event requiring containment isolation.

The design of the EFS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 as related to containment isolation. (Reference 9A.9.3-1).

#### **9A.9.3.2 Safety Design Bases**

As part of Defense Line 3 the EFS provides containment isolation valves on piping that penetrates the containment boundary.

#### **9A.9.3.3 Description**

Refer to Figure 9A.9.3-1a which depicts a typical drain sump and Figure 9A.9.3-1b which depicts the containment drain sump.

The EFS consists of the drainpipes, collection sumps, sump pumps, interconnecting piping, and instrumentation to provide for the collection and removal of liquid waste in the plant. The EFS is provided for the Reactor Building, Turbine Building, Radwaste Building and PLSA.

#### **General Floor Drains and Sumps**

All liquid waste collected in the radioactive drains are treated as High Conductivity Waste (HCW). Liquid waste is floor drain wastes, equipment drains, and process drains collected throughout the entire facility. These wastes are collected by drains and drain headers, routed by gravity drainage piping to a sump, and pumped to LWM for processing. All liquid waste is considered HCW in the

BWRX-300 with the exception of a small subset of potentially clean drains and sumps that are used to keep chemicals from mixing with radwaste and to reduce the amount of waste sent to processing. Liquid waste testing is utilized in order to test for radioactive contamination among the potentially clean collection samples.

The EFS continuously collects the liquid waste that is generated throughout the plant during all modes of plant operation. The sumps receive and store the liquid wastes until the contents are pumped to the LWM. Each sump has a cover to prevent debris from entering the sump. The sumps are vented to their associated HVAC exhaust systems for control of airborne contamination. The sumps are also provided with flush capability to minimize the solid deposit within the sump and transfer piping. Clean flush water can be added to clean out a sump prior to maintenance activities.

Open drainage lines from areas that are required to maintain an air pressure differential but drain to the system are provided with a water seal.

### **Oily Wastes Drains and Sump**

Dedicated oily waste collection sumps are provided at the base of the EHC unit, lube oil skid and diesel generators in case of a leak. A temporary pump is required to pump oily waste from the sump pit to a collection barrel for transport to the SWM (Chapter 11, Subsection 11.4) drum evaporator for de-watering and disposal.

### **Pressurized Containment Sump**

Three divisionally separated sumps are included in the basement floor of the RB. One of the sumps is a pressurized containment sump. The EFS has drain piping in the SCCV floor to drain any leakage or condensation collected in containment to the pressurized sump. The pressurized sump is also used for containment leak detection and measurement. The pressurized sump's upstream piping penetrates the SCCV which requires the use of redundant isolation valves to ensure containment isolation. The isolation valves fail closed and automatically close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Section 7.3).

#### ***9A.9.3.3.1 Component Description***

### **Sump Pumps**

The sump pumps are vertical, centrifugal types, driven by electric motors. Two parallel 100% capacity pumps (run pump and standby pump) are provided for each sump.

Maintenance valves and flanged connections are provided for sump pumps so that they can be isolated and removed from the sumps for maintenance and repair. Check valves in the pump discharge piping prevent backflow of waste into the sump.

### **Collection Sumps**

The sumps are sized to accommodate the normal anticipated daily inputs without overflow in conjunction with operation of one sump pump.

Collection sumps are lined cavities in the floor slab of their respective building. The sumps have tight fitting, but not gastight, steel plate covers to prevent the entrance of debris as well as to minimize airborne contamination. The sumps have vents that are piped to the building HVAC interfacing exhaust. The sumps are provided with grab sampling connections.

The pressurized containment sump is designed in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Division 1 (Reference 9A.9.3-3). The pressurized sump vessel is steel

lined and shielded due to its interface with the primary containment. Any leakage from the pressurized sump is contained in the lined cavity, eliminating direct contact with the bare concrete.

A pressurized vessel is used in place of a traditional sump in the Reactor Building basement due to its open interface with primary containment during normal operation. The pressurized sump is located outside containment to maintain a dry containment. The pressurized vessel provides an interface for a drain line, flush water, level transmitters and pump connections.

### **Piping and Valves**

All piping is in accordance with the requirements of ASME B31.1 (Reference 9A.9.3-4) except for the containment penetration. As part of Defense Line 3, the EFS provides containment isolation valves on piping that penetrates the containment boundary. Accordingly, the piping penetrating containment, up to and including both isolation valves, is Safety-Class 1.

The EFS piping penetrating containment is an extension of the containment boundary and is designed to Category A; ASME BPVC Section III, Division 1, Class 2 requirements. The arrangement of the isolation valves and connecting piping is such that a single active failure or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the EFS containment penetrations. The containment isolation valves are classified as Seismic Category B.

The floor drain line in containment connects directly to the containment atmosphere, penetrates the primary reactor containment, and is provided with containment isolation valves. There are two containment isolation valves placed in series, located outside the containment vessel, and placed as close to the primary containment wall as practical.

These isolation valves are located to provide accessibility for maintenance, inspection, and testing during all modes of reactor operation.

The pressurized sump system is credited for unidentified leak measurement, so the isolation valves on the EFS containment drain line are designed to be normally open to connect the drain line to the pressurized sump. To ensure a power source availability for the containment isolation valves, the isolation valves are supplied by different power source divisions. The EFS containment isolation valves have provisions to close automatically. Upon loss of actuating power, the automatic isolation valves are designed to fail closed, taking the position that provides greater safety.

All ESF piping is stainless steel to minimize corrosion.

Piping entering the collection sumps is turned down and terminated below the lowest sump fluid level to which the sump pump can draw. This provides a water seal to prevent gas flow and cross-contamination of building areas.

#### **9A.9.3.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing corrodents and mitigating the degradation of materials through material chemistry, heat treatment, material processes controls and periodic inspection. The ESF piping is manufactured from stainless steel and the sumps are manufactured using materials which have a high degree of resistance to effluents treated by the system. Proper selection of radiation-resistant materials of construction is included in individual equipment specifications as all liquid collected is treated as potentially contaminated.

#### **9A.9.3.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.9.3-1 for ESF interfaces.



#### **9A.9.3.6 System and Equipment Operation**

##### **Normal Operational Concept**

The ESF operates automatically in the same manner under all normal operating modes of the plant (startup, power operation, hot shutdown, cold shutdown, and refueling). Liquid wastes generated in the plant are directed into drain fixtures and drain piping. The ESF continually collects these radioactive or potentially radioactive liquid wastes and convey the wastes via gravity drainage piping systems to their respective collection sumps. Each pump is sized to accommodate the normal anticipated daily inputs into the sump. The system is not designed to accommodate fire suppression water. However, the system design features prevent drain or flood water from backing up in the drainage system into areas housing Safety-Category function SSC's.

No credit is taken for the drainage system in the internal flooding analysis.

Dual 100% capacity sump pumps are provided on each sump pump to pump liquid waste to LWM. The second pump automatically starts on sump high-high water level, provides redundancy in the event of sump pump failure, and assists the single pump when abnormally high inputs into the sump occur.

##### **Off-Normal Operational Concept**

In the event of a Loss-of-Coolant Accident (LOCA) signal, the containment isolation valves perform a Safety-Category function and close, thereby maintaining the integrity of the SCCV.

Under Design Basis Event (DBE) conditions and assuming the unavailability of normal power, the ESF does not operate, nor is it required to be functional. The containment isolation valves fail closed.

All drainage piping whose collapse could result in a loss of function of Safety-Category function SSC equipment, is seismically analyzed to remain intact following an SSE.

#### **9A.9.3.7 Instrument and Control**

##### **Instrumentation**

Redundant level sensors and transmitters are provided for each sump. Redundancy is appropriate for ensuring reliability and availability of the I&C systems. Level signals are used to start and stop the sump pumps based on High and Low setpoints. The level transmitters on the pressurized containment sump are also used to detect, measure, and trend leakage inside containment. Pump run timers are provided for each sump pump.

Local pressure transmitters are provided in the discharge piping of each sump pump which display the pressure remotely in the Main Control Room. Additionally, the pressurized sump that penetrates the SCCV is provided with discharge flow transmitters that displays the flow remotely in the Main Control Room.

Isolation diaphragms are installed between all gauges or transmitters and the piping, to avoid contaminating the instrument.

##### **Controls**

The following describes the key control features of the EFS.

Sump pump operations automatically maintain the sump level. The lead sump pump starts upon a high level indication and stops on low level indication. When the capacity of one sump pump is exceeded and the sump level rises above the high level, a high-high level setting is reached, and the standby pump starts. Both pumps operate until the low level is reached, causing both pumps

to stop. Alarms are annunciated at the high-high level to notify the operator of potential sump overflow and at the low-low level to notify the operator of potential air cross-contamination due to the loss of water seal. Sump pump controls are in the Main Control Room.

Where needed, sumps are provided with remote flush capability to clean out sumps prior to maintenance activities to minimize solid deposits.

The EFS containment isolation valves are controlled via the Safety Class 1 Instrumentation and Control System, and close upon receiving an isolation signal. These valves can also be remotely operated from the control rooms.

#### **9A.9.3.8 Monitoring, Inspection, Testing, and Maintenance**

The containment isolation valve closure time is measured during the valve operability test and the leakage is measured during the valve leakage test as specified in the containment leakage testing procedures. Each sump in the EFS utilizes recirculation lines with a grab sample tap to test/analyze the contents of the sump. Leak detection and inspection for primary containment isolation features comply with the requirements of ASME Boiler and Pressure Vessel Code Section III, Division 1, Class 2 (Reference 9A.9.3-2). A test connection is provided between the isolation valves to support local leak rate testing of the primary containment boundary.

#### **9A.9.3.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.3.10 Performance and Safety Evaluation**

As part of DL3, the EFS performs a containment isolation Safety-Category function. The CIVs are designed to close upon receipt of an isolation signal from the Safety Class 1 Instrument and Control System. The CIVs are designed to maintain the leak tightness of the containment in the event of an accident and prevent radioactive releases to the environment that exceed prescribed limits.

#### **9A.9.3.11 References**

- 9A.9.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.3-2 ASME Boiler and Pressure Vessel Code (BPVC), Section III, "Rules for Construction of Nuclear Power Plant Components, Division 1 - Subsection NC, Class 2 Components," American Society of Mechanical Engineers.
- 9A.9.3-3 ASME Boiler and Pressure Vessel Code (BPVC), Section VIII, "Rules for Construction of Pressure Vessels, Division 1," American Society of Mechanical Engineers.
- 9A.9.3-4 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

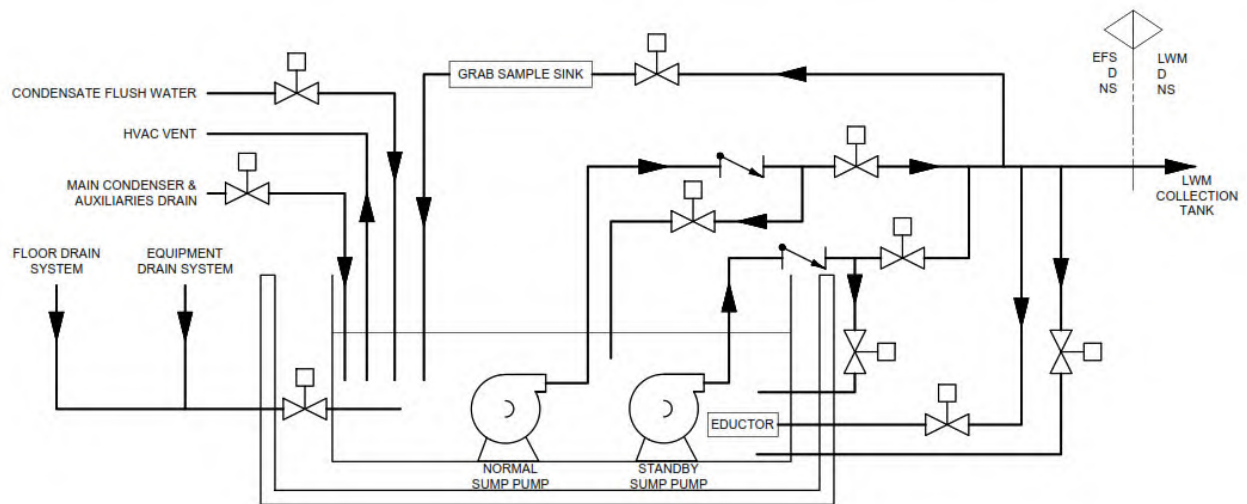
**Table 9A.9.3-1: Equipment and Floor Drain System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Safety Class 1 Instrumentation and Control System	Provides all Safety Class 1 I&C control (Containment Isolation Valves)	Primary Containment Isolation Valves
Non-safety Instrumentation and Control System	Provides all Non-Safety Category I&C control. Sump pump logics, valve logics, instrumentation (pressure, flow, temperature, level) etc.	Instruments
Process Radiation and Environmental Monitoring System	Provides process sampling for radiological analysis radiation detection and the instrumentation used for containment leak detection and monitoring	Sampling locations
Control Rod Drive System	Provides drain piping for CRD pumps and the Hydraulic Control Unit room. The CRD wash down drains into the pressurized containment sump	Drain hub
Liquid Waste Management	Sump pumps discharge effluent to LWM for processing (LWM collection tank) Provides seal water and flushing water for sumps	Collection Tank
Solid Waste Management	Oily waste is collected and pumped to SWM drum evaporator for dewatering and disposal	Drum Evaporator
Condensate and Feedwater Heating	Provides drain piping from CFS to EFS	Drain hub
Condensate Filters and Demineralizers	Provides drain piping from CFD to EFS	Drain hub
Main Turbine Equipment	Provides drain piping from MTE to EFS	Drain hub
Main Condenser and Auxiliaries	Provides drain piping from MCA to EFS	Drain hub
Plant Cooling Water	Dedicated clean sump for PCW system. The PCW dedicated sump discharges the collected water back to the makeup water surge tank	Drain hub
Plant Pneumatics System	Provides air to EFS Air Operated Valves	PPS Tank/Air Operated Valves

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

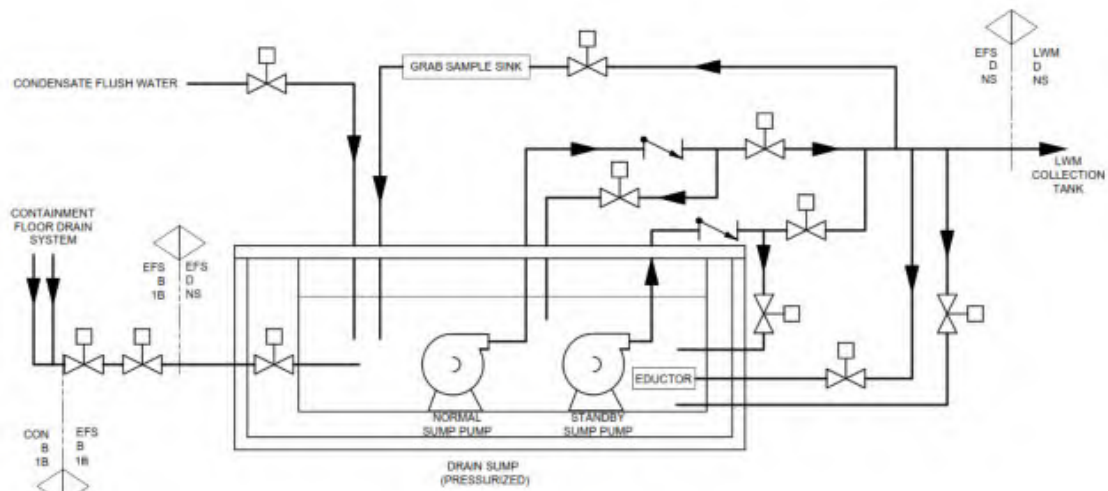
Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Electrical Distribution System	Provides diesel generator oil catching from Safety Class 2 and 3 Electrical Distribution System	Drain Hub
Non-Safety Electrical Distribution System	Provides power to pumps and valves	Pump motors and valve motors
Steel-plate Construction Containment Vessel	EFS provides piping to drain the SCCV	SCCV wall and piping penetration
Containment Cooling System	Pressurized containment sump collects condensate from the air handling unit's drain pans	Air Handling Units drain pan
Heating Ventilation and Cooling System	The collection sumps are vented to Heating, Ventilation and Air Conditioning ductwork	Sump vent duct
Fire Protection System	EFS collects FPS water EFS sump overflow due to Fire Protection System water is allowed with a maximum flood depth required in all associated buildings	Collection Sumps
Reactor Building Structure	Sloped drain piping	Reactor Building structure and floor
Turbine Building Structure	Sloped drain piping embedded in concrete floor	Slab on grade concrete floor
Radwaste Building Structure	Sloped drain piping embedded in concrete floor	Slab on grade concrete floor
Yard/Balance of Plant	EFS discharges PPS condensate from the PPS to the yard storm water drain	Drain hub

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 9A.9.3-1a: Floor Drain System Drain Sump (Typical)**

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION



**Figure 9A.9.3-1b: Floor Drain System Containment Drain Sump (Pressurized)**

#### **9A.9.4 Interfacing Water Systems**

##### **9A.9.4.1 Potable Water System**

The Potable Water System design is dependent on the site-specific water pathways. The system is designed to supply the required volume of potable water during peak demand periods.

##### **9A.9.4.2 System and Equipment Functions**

###### **Normal Functions (Non-Safety-Category)**

Provides for potable water distribution inside the power block. The Potable Water System brings potable water to the power block from the city water source.

###### **Normal Function (Safety Category)**

There are no Safety-Category functions related to the Potable Water System.

###### **Off-Normal Functions (Non-Safety Category)**

There are no specific off-normal functions related to the Potable Water System.

###### **Off-Normal Functions (Safety Category)**

There are no Safety-Category functions related to the Potable Water System.

The safety classification of the Potable Water System is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.9.4.12-1).

##### **9A.9.4.3 Safety Design Bases**

The Potable Water System does not perform, ensure, or support any Safety-Category function, and thus has no safety design bases.

##### **9A.9.4.4 Description**

The potable water system provides potable water to the power block. The system includes faucets, toilets, showers, and hot water heaters.

The potable water system provides makeup water to the demineralized water trailers if needed.

The Potable Water System consists of potable water supply piping from the Municipality of Durham. The supply line includes isolation valves, flow totalizer and instrumentation. Heat tracing is provided for all outside above ground piping and instrumentation. A distribution header supplies the power block and other onsite buildings. Provisions are made in the distribution header to accommodate future demand.

Potable water supplies the following areas:

1. Mobile demineralized water trailers as required to supplement Darlington Nuclear Generating Station (DNGS) demineralized water supply
2. Restrooms, food service equipment and battery room safety showers in the Control Building
3. Water is supplied to the toilet in the Main Control Room

###### **9A.9.4.4.1 Component Description**

The Potable Water System components include: underground supply lines, isolation valves, and necessary instrumentation. Heat tracing is provided for all outside above ground piping and

instrumentation. Where required, back flow preventers are installed. Hot water heaters, faucets and showers are installed where needed.

Piping and valves associated with the Potable Water System are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include but are not limited to ASME B31.1 Power Piping (Reference 9A.9.4.12-2), National Plumbing Code of Canada (Reference 9A.9.4.12-3), Regulation 332/12 Ontario Building Code (Reference 9A.9.4.12-4), and American Society for Testing and Materials D3350-14 Standard Specification for Polyethylene Plastics Pipe and Fittings Material (Reference 9A.9.4.12-5).

#### **9A.9.4.5 Materials**

The Potable Water System provides piping, valves, and other control components to distribute potable water to final use locations. Potable water system component materials are built to the Canadian National Building Code.

#### **9A.9.4.6 Interfaces with Other Equipment or Systems**

The potable water system interfaces with the following other systems:

Demin Water – There is a potable water connection to provide for temporary potable water supply to portable demineralizer trailers.

Sanitary Sewer – Potable water supplies water to toilets, sinks, showers which drain to the sanitary sewer system.

Electrical – Low voltage power from the Non-Safety Electrical Distribution System is supplied to components in the system to provide for heating of water.

#### **9A.9.4.7 System and Equipment Operation**

The potable water system relies on system pressure being supplied from the Municipality of Durham. Hot water heaters are provided where necessary. The hot water heaters temperature is automatically controlled.

The Potable Water System is designed for operation in all modes.

#### **9A.9.4.8 Instrumentation and Control**

Instrumentation for the potable water system includes flow metres to track usage of water, back flow preventers where required by code, temperature control thermostats in the hot water heaters, and relief valves in the system where required.

Instrumentation such as valve position switches, pressure gauges, and temperature sensors are located throughout the system.

#### **9A.9.4.9 Monitoring, Inspection, Testing, and Maintenance**

The potable water system is tested and commissioned in accordance with the National Plumbing Code of Canada and Ontario's Watermain Disinfection Procedure. Commissioning includes hydrostatically tested for leak-tightness, disinfection, and flushing of the system.

Periodic testing for residual chlorine and micro biologics is done in accordance with the Safe Drinking Water Act, the Ontario Drinking Water Quality Management Standard, and Guidelines for Canadian Drinking Water Quality.



#### **9A.9.4.10 Radiological Aspects**

The Potable Water System has no interconnections to systems with the potential for containing radioactive material.

Chapter 12, Subsection 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.4.11 Performance and Safety Evaluation**

There is no potable water piping in the reactor building. Potable water is limited to Non-Safety-Category function areas of the plant. The MCR washroom and kitchen area sinks, and potable water piping are sufficiently remote from the control system to eliminate the potential for interactions.

#### **9A.9.4.12 References**

- 9A.9.4.12-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.4.12-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.9.4.12-3 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.4.12-4 O. Reg. 332/12, "Building Code," Government of Ontario.
- 9A.9.4.12-5 American Society for Testing and Materials D3350-14, "Standard Specification for Polyethylene Plastics Pipe and Fittings Material."

### **9A.9.5 Makeup Water System**

The Makeup Water System consists of supply piping from the Darlington Nuclear Generating Station (DN GS) site, trailer hookup piping, storage tank, transfer pumps, and distribution piping.

#### **9A.9.5.1 System and Equipment Functions**

##### **Normal Functions (Non-Safety-Category)**

Provides for demineralized water storage and distribution.

##### **Normal Function (Safety Category)**

The system does not perform any Safety-Category functions during normal conditions.

##### **Off-Normal Functions (Non-Safety Category)**

Following a reactor trip with no condenser available, the Makeup Water System provides makeup to the Isolation Condenser pools (Chapter 5, Subsection 5.8).

##### **Off-Normal Functions (Safety Category)**

The Makeup Water System provides containment isolation of Makeup Water System piping penetrating the SCCV. This isolation is provided by two locked closed manual isolation valves which are Safety Class 1 and Defense Line 3 – one valve inboard and one valve outboard of the Primary Containment. The containment isolation valves are only open during Mode 5 or 6.

The Makeup Water System is not required to operate during or after a design basis event except for performing the containment isolation function. The design of the Makeup Water System meets

CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 as related to containment isolation. (Reference 9A.9.5.11-1).

#### **9A.9.5.2 Safety Design Bases**

The Makeup Water System containment penetration and isolation valves perform the safety function of maintaining containment integrity.

#### **9A.9.5.3 Description**

The makeup water supply from the DNGS site is sized to provide for normal makeup water needs for the unit as well as additional capacity to refill the ICS pools in a reasonable timeframe following a typical initiation. During certain shutdown/refueling/startup mode evolutions the increases in plant water consumption may require use of a temporary demineralization subsystem to be used as a supplemental water source.

Demineralized water of the proper water quality specification is supplied to the single demineralized water storage tank located in the site yard. The tank is heated to prevent freezing. Two 100% pumps supply the demineralized water system.

The demineralized water system provides makeup water to: Condensate Storage Tank, Isolation Condenser Pools, Ventilation Humidification, Chilled Water and Plant Cooling Water surge tanks, Boron Injection Tank, and washdown stations on the refuel floor and in containment. Additional demineralized water connections are located where needed for maintenance of systems.

The makeup water equipment and associated piping in contact with demineralized water are fabricated from corrosion-resistant materials to prevent contamination of the makeup water.

Piping and valves associated with the Makeup Water System are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include but are not limited to ASME B31.1 Power Piping (Reference 9A.9.5.12-2), National Plumbing Code of Canada (Reference 9A.9.5.12-3) – American Water Works Association D100 (Reference 9A.9.5.12-4).

#### **9A.9.5.4 Materials**

Water quality requirements are used in the selection of Makeup Water System components.

#### **9A.9.5.5 Interfaces with Other Equipment or Systems**

Refer to Table 9A.9.5-1 for Makeup Water System interfaces with other systems.

#### **9A.9.5.6 System and Equipment Operation**

The demineralized water storage tank is filled as necessary from the DNGS water treatment plant. Filling of the tanks is a batch process that is manually controlled. Level alarms prompt the operator to secure filling when needed.

Normal plant operation includes Power Operation, Startup Operation, Hot Shutdown Operation, Stable Shutdown Operation, Cold Shutdown Operation, and Refueling Operation. During normal operations the Makeup Water System provides demineralized water storage and distribution to required systems and equipment in the plant.

The demineralized water in containment is only used when the plant is shutdown. During plant operation both inside and outside manual containment isolation valves are locked closed.

#### **9A.9.5.7 Instrumentation and Control**

Level alarms on the demineralized water storage tank alert the operator to high or low level situations.

The storage tank heater as well as heat trace on above ground yard piping is automatically controlled.

#### **9A.9.5.8 Monitoring, Inspection, Testing, and Maintenance**

Containment Isolation Valves are periodically tested to validate operability and determine if valve leakage is within acceptable limits. Test and vent connections are provided at the containment isolation valves to verify that the valves meet the local leak rate limits.

Makeup water is monitored and tested to ensure it meets the water quality requirements for interfacing systems.

Periodic and condition-based maintenance are completed for instrumentation, pump vibrations, and other equipment to ensure the proper performance of the system.

#### **9A.9.5.9 Radiological Aspects**

Chapter 12, Subsection 12.3 provides information pertaining to design measures that can be taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are as ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.5.10 Performance and Safety Evaluation**

The Makeup Water System performs the containment isolation Safety-Category function using two manual valves. The containment isolation valve arrangement meets the requirements of Section 8.6.6 of REGDOC-2.5.2 (Reference 9A.9.5-1).

#### **9A.9.5.11 References**

- 9A.9.5-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.5.11-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.9.5.11-3 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.5.11-4 AWWA D100, "Welded Carbon Steel Tanks for Water Storage," American Water Works Association.

NEDO-33958 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 9A.9.5-1: Makeup Water System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Control System	Provides all SCN Instrumentation and Control for the demineralized water equipment, potable water, sewage treatment, and hydrogen subsystems	Non-Safety Class Instruments
Non-Safety Electrical Distribution System	Provides power to demineralized water transfer pumps and tank/piping heating	SCN Pumps and heater, and controls
Process Radiation and Environmental Monitoring System	Provides demineralized water to Process and Radiation Monitoring for process sampling	Sampling points and RWB chemical Laboratory
ICS Pool Cooling and Cleanup System	Provides Demineralized water to the IC pools through the ICS Cooling and Cleanup system	Interface valve
Boron Injection System	Provides demineralized water to the Boron Injection Sodium Pentaborate Storage Tank for filling	Interface valve
Plant Cooling Water System	Provides demineralized water to the surge tanks	Interface valve
Chilled Water Equipment System	Provides demineralized water to the glycol auto fill unit	Interface valve
Heating, Ventilation and Cooling System	Provides demineralized water to CB AHUs used for space humidification	AHUs
Liquid Waste Management System	Provides demineralized water to the Condensate storage tank	Interface valve
Refueling Equipment and Servicing	Provides demineralized water connections on the refueling floor for use during outages	Interface valve
Main Condenser and Auxiliaries	Provides fill and makeup water to the condenser vacuum skids for seal water separators	Interface valve
DNGS Demineralized Water System	System is supplied demineralized water from the DNGS Demineralized Water System	Yard demineralized water tie-point connection

## **9A.9.6 Sanitary Water Systems**

### **9A.9.6.1 System and Equipment Functions**

#### **Normal Functions (Non-Safety Category)**

Provides for sewage water collection and delivery to the existing DNGS East Sewage Lift Station.

#### **Normal Function (Safety Category)**

There are no normal Safety-Category functions related to the Sanitary Water System.

#### **Off-Normal Functions (Non-Safety Category)**

There are no off-normal functions related to the Sanitary Water System.

#### **Off-Normal Functions (Safety Category)**

There are no specific off-normal functions related to the Sanitary Water System.

The safety classification of the Sanitary Waste System is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.9.6.11-1).

### **9A.9.6.2 Safety Design Bases**

The Sanitary Water System does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

### **9A.9.6.3 Description**

The Sanitary Sewer Collection and Delivery Subsystem collects sewage from the facility washrooms and break areas and transfers the sewage to the existing DNGS East Sewage Lift Station. The sewage system is a non-radiologically contaminated system and collects sewage only from areas outside of any radiologically controlled areas in the CB. This subsystem is sized for normal plant operation of the CB. The Administration Building and the Security Building sewage system is provided by OPG.

The Sanitary Sewage Handling Subsystem is designed to prevent raw sewage overflow in the event of a power outage.

Piping and valves are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include National Plumbing Code of Canada (Reference 9A.9.6.11-2), and Regulation 332/12 Ontario Building Code (Reference 9A.9.6.11-3).

### **9A.9.6.4 Materials**

The water quality requirements are used in the material selection and design of the water treatment systems.

### **9A.9.6.5 Interfaces with Other Equipment or Systems**

The sanitary water system interfaces with the DNGS waste treatment system.

### **9A.9.6.6 System and Equipment Operation**

The Sanitary Water System is designed for operation in all modes.

Sanitary waste drains are gravity drained through underground piping to a lift station which is part of the DNGS waste treatment system.

#### **9A.9.6.7 Instrumentation and Control**

Alarm indication is provided as needed to inform the operator of the status of the DNGS East Lift Station pump.

#### **9A.9.6.8 Monitoring, Inspection, Testing, and Maintenance**

Inspection and testing of the Sanitary Water System is performed in accordance with the applicable requirements of the Ontario Building Code.

#### **9A.9.6.9 Radiological Aspects**

The Sanitary Water System is a non-radiologically contaminated system and only collects sanitary waste from areas outside of any radiologically controlled areas in the CB.

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.6.10 Performance and Safety Evaluation**

The Sanitary Water System does not perform, ensure, or support any Safety-Category function; therefore a Safety Evaluation is not required.

#### **9A.9.6.11 References**

- 9A.9.6.11-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.6.11-2 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.6.11-3 O. Reg. 332/12, "Building Code," Government of Ontario.

## **9A.9.7 Chemistry Systems**

### **9A.9.7.1 Primary Coolant**

Not Applicable, the terminology "Primary Coolant" refers to a Pressurized Water Reactor.

### **9A.9.7.2 Secondary Coolant**

Not Applicable, the terminology "Secondary Coolant" refers to a Pressurized Water Reactor.

## **9A.9.8 Other Process Media and Other Materials**

### **9A.9.8.1 Hydrogen Water Chemistry**

#### **9A.9.8.1.1 *System and Equipment Functions***

The Hydrogen Water Chemistry System adds hydrogen into the feedwater system at the Reactor Feed Pump suction and oxygen (as a constituent of air) into the Offgas System (Chapter 11, Section 11.3) via the Service Air System (Subsection 9A.4.1). The Hydrogen Water Chemistry System is included in the BWRX-300 design for the purpose of reducing and mitigating IGSCC in reactor vessel internals.

#### **9A.9.8.1.2 *Safety Design Bases***

The Hydrogen Water Chemistry System does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

#### **9A.9.8.1.3 *Description***

The Hydrogen Water Chemistry System provides for an established technique for reducing and preventing the growth rates of Intergranular Stress Corrosion Cracking (IGSCC) in reactor vessel internals. The mitigation of IGSCC is achieved from the reduction of oxygen and other oxidizing species (oxidants) in the reactor coolant. This is accomplished by injecting hydrogen into the feedwater at the feedwater pump suction. The injected hydrogen suppresses the radiolytic formation of oxidants in the reactor core. To compensate for any excess hydrogen which may travel downstream and be removed from the main condenser by the Offgas System, a corresponding amount of oxygen, as a constituent of injected air provided by the Service Air System, is injected into the Offgas System prior to Offgas Recombiner.

Reduction of oxidants results in a reduction of the Electrochemical Corrosion Potential, which is the measurement that is used to predict initiation and growth of IGSCC. When the Electrochemical Corrosion Potential is reduced to an acceptable level, IGSCC crack initiation stops, and crack growth rates are minimized. The BWRX-300 design employs NobleChem™ technology as a means to provide for the injection of noble metal(s) into the reactor to aid in the protection of reactor vessel internals from IGSCC in combination with the addition of hydrogen by the Hydrogen Water Chemistry System. As a result of NobleChem™ injection, the noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimize the oxygen available to initiate and encourage IGSCC crack growth. Refer to Subsection 9A.9.9 for a description of the application of NobleChem™ technology.

The Hydrogen Water Chemistry System injects hydrogen into the feedwater system at the Reactor Feed Pump (RFP) suction to mitigate IGSCC in the reactor internals. The injected hydrogen causes a reduction in dissolved oxygen and other oxidants within the reactor internals by lowering the radiolytic production of hydrogen and oxygen in the vessel core region.

Hydrogen addition to the feedwater results in an excess ratio of hydrogen to oxygen at the entrance to the Offgas System. The Hydrogen Water Chemistry System injects a stoichiometric

amount of oxygen (as a constituent of air) upstream of the offgas recombiner to combine with the hydrogen reaching the recombiner. The amount of air (oxygen) can be determined from the hydrogen injection rate and is controlled by the Hydrogen Water Chemistry System utilizing time delays in the controls to ensure that hydrogen and oxygen reach the offgas recombiner in balanced amounts. The net result is that with air (oxygen) injection, the Offgas System operates at conditions (relative to gas constituents needed to recombine to form water vapor) very much like the conditions without Hydrogen Water Chemistry System. For most BWRs that use NobleChem™ technology with a Hydrogen Water Chemistry System featured with air injection into the offgas system, the actual air flow injection can be determined from the hydrogen injection rate and condenser air inleakage and, is controlled by the Hydrogen Water Chemistry System controller to maintain sufficient percent oxygen exit downstream of the offgas recombiner.

The Hydrogen Water Chemistry System consists of a Main Control Panel, a Hydrogen Isolation Module (to isolate the hydrogen supply from the power plant buildings if necessary), a Hydrogen Flow Control Module, which controls the flow of hydrogen into the Feedwater System, and an Air Flow Control Module, which controls air flow to the Offgas System. Note that Hydrogen Area Monitors are also included in the system which monitor for local hydrogen in the areas of susceptible system equipment.

#### **9A.9.8.1.4 System and Equipment Operation**

##### **Power Operation**

The Hydrogen Water Chemistry System can only be operated during power operations. The Hydrogen Water Chemistry System begins operation when the RFP pump is in service and Steam Jet Air Ejectors (SJAE) and Offgas System are in-service during plant startup.

#### **9A.9.8.1.5 Instrumentation and Control**

Instrumentation is provided to control the injection of hydrogen and the injection of oxygen (as a constituent of air) via the Service Air System. Automatic control features in the Hydrogen Water Chemistry System minimize the need for operator attention and improves performance.

#### **9A.9.8.1.6 Monitoring, Inspection, Testing, and Maintenance**

The Hydrogen Water Chemistry System is demonstrated functional by its use during normal operation.

#### **9A.9.8.1.7 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.8.1.8 Performance and Safety Evaluation**

The Hydrogen Water Chemistry System does not perform a Safety-Category function. However, the Hydrogen Water Chemistry System is used, along with other measures, to reduce the likelihood of corrosion failures that would adversely affect plant availability. The means for storing and handling hydrogen utilizes the guidelines in Electric Power Research Institute (EPRI) Report "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" (Reference 9A.9.8-1).

#### **9A.9.8.1.9 References**

- 9A.9.8-1 EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," Electric Power Research Institute.



### **9A.9.9 NobleChem™ Injection**

#### **9A.9.9.1 System and Equipment Functions**

On-Line NobleChem™ (OLNC) provides a means for the injection of noble metal salt solution directly into the reactor coolant flow path. The purpose of injecting noble metal into the reactor is to aid in the protection of reactor vessel internals from IGSCC in combination with the addition of hydrogen by the Hydrogen Water Chemistry System (Subsection 9A.9.8.1).

#### **9A.9.9.2 Safety Design Bases**

The OLNC does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

#### **9A.9.9.3 Description**

The OLNC application process deposits noble metal salt solution directly into the reactor coolant flow path. The resulting noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimize the oxygen available to initiate and encourage IGSCC crack growth. Due to the catalyst effect from noble metal loading, utilizing OLNC in conjunction with a Hydrogen Water Chemistry System mitigation strategy results in a much lower amount of injected hydrogen required to achieve IGSCC mitigation levels than the use of Hydrogen Water Chemistry System alone. This need for less hydrogen to provide mitigation results in lower main steam line dose rates.

OLNC operates in conjunction with the Hydrogen Water Chemistry System although they are both separate systems which have no mechanical or electronic interrelationships. The Hydrogen Water Chemistry System is used to provide a ratio of hydrogen to oxygen in excess of the stoichiometric balance which causes a significant decrease in the electrochemical corrosion potential of the reactor vessel and internal components. As long as excess hydrogen is maintained, crack initiation and growth are greatly reduced, even at high bulk liquid oxidant levels. OLNC injection provides a catalytic surface on reactor vessel internals to enhance the function of the Hydrogen Water Chemistry System. Based on the catalytic surface recombination efficiency provided by OLNC, less hydrogen is required (from the Hydrogen Water Chemistry System) to establish the Electrochemical Corrosion Potential to mitigate IGSCC initiation and significantly reduce IGSCC crack growth rates.

The OLNC injection skid is installed as permanent plant equipment outside of the Steel-plate Composite Containment Vessel in a Non-Safety-Category area as close as possible to the Feedwater injection tap(s). A direct connection to a supply of demineralized water provides the carrier flow to lessen the residence time of the noble metal solution in the injection line. The effluent from the OLNC injection skid is injected into a tap (or taps) in the Feedwater System (via an injection quill) at a point that lessens the loss of noble metal prior to entering the reactor.

Once the flow from the injection skid reaches the Feedwater System, the process employs the reactor coolant as the transport medium to deposit noble metal on the surface of all wetted reactor components and inside existing cracks.

The boundary between OLNC and the Condensate and Feedwater Heating System (Chapter 10, Subsection 10.3.2) is downstream of the isolation and root valves to the CFS to a shared flange with the CFS (and also includes the injection quill). A single input tie for demineralized water as well as plant electrical power from the Non-Safety Electrical Distribution System is provided to support operation of the system.

#### **9A.9.9.4 System and Equipment Operation**

##### **Power Operation**

The OLNC injection is designed to be performed during power operations.

#### **9A.9.9.5 Instrumentation and Control**

Instrumentation is provided to monitor and control the injection of a noble metal solution into the reactor coolant flow path at the injection skid.

#### **9A.9.9.6 Monitoring, Inspection, Testing, and Maintenance**

OLNC injection is demonstrated functional by its use during normal operation.

#### **9A.9.9.7 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

#### **9A.9.9.8 Performance and Safety Evaluation**

OLNC does not perform a Safety-Category function. The OLNC Injection System is used, along with the Hydrogen Water Chemistry System to reduce the likelihood of IGSCC.

#### **9A.9.10 Chemical Bases of Water Treatment**

Refer to Chapter 13, Subsection 13.3.1 for information pertaining to the Chemistry Control Program.

#### **9A.10 Storage System for Non-Permanent Equipment Used in Design Extension Conditions**

The storage system for non-permanent equipment used in Design Extension Conditions is developed based upon Emergency Mitigating Equipment requirements. The design of the storage system takes into consideration the aggregate set of on-site and off-site resource considerations for the hazards that are applicable to the site as well as final BWRX-300 design considerations. These factors plus consideration for the following are used as design inputs to size, locate, deploy, and operate the storage system”

- Protection of EME
- Deployment of EME
- Procedural Interfaces
- Utilization of off-site resources
- Use of existing site facilities for storage



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33959

Revision 2

March 7, 2023

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 9B  
Civil Engineering Works and Structures**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	Section 9B.4	Reference correction per customer acceptance review.
2	Acronym list	Updated to include added acronyms.
2	Table of Content, List of Tables and List of Figures	Updated.
2	Section 9B.1	Added reference to Section 9B.3.
2	Section 9B.1.1	Edited to remove reference to RB walls.
2	Subsection 9B.1.2	Added reference to Table 3.5-1 and 3.5.2. Edited to include reference to Chapter 2, Subsection 2.7.4.7.6.
2	Subsection 9B.1.3	Added reference to Figure 9B-1 and details on mat foundation. Added wing walls to lateral load resisting systems.
2	Subsection 9B.1.5	Edited to include reference to integrated RB structures.
2	Subsection 9B.1.6	Edited for clarity on structural design of the common mat foundation.
2	Subsection 9B.1.9	Edited to include reference to Chapter 2, Subsection 2.11.3.
2	Section 9B.2	Edited to include RB exterior wall terminology.
2	Subsection 9B.2.1.2	Added flooding to list of internal hazards and reference to Chapter 3, Section 3.3 and Chapter 6, Subsection 6.3.5.
2	Subsection 9B.2.1.3	Added reference to Chapter 6, Subsection 6.3.4.
2	Subsection 9B.2.1.4	Edited to include reference to design specifications and removed details on concrete used for shielding purposes.
2	Subsection 9B.2.1.4.1	Modified heading for Subsection 9B.2.1.4.1.
2	Subsection 9B.2.1.4.2	Updated steel faceplates properties and references consistent with 9B-4.
2	Subsection 9B.2.1.4.3	Updated references consistent with 9B-4.
2	Subsection 9B.2.1.4.4	Updated references consistent with 9B-4.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
2	Subsection 9B.2.1.5	Removed the word “accurately” and updated reference consistent with 9B-4.
2	Subsection 9B.2.1.8	Edited details in reference to Chapter 3, Subsection 3.5.5.6.
2	Subsection 9B.2.1.9	Added details and reference to Subsection 9B.2.2.3 and Figure 9B-1.
2	Subsection 9B.2.1.10	Removed the word “expected” and updated reference consistent with 9B-4.
	Subsection 9B.2.1.10.1	Updated references consistent with 9B-4.
2	Subsection 9B.2.1.10.2	Added details in reference to Appendix 9B.A and Appendix 9B.E. Removed reference to Open Item.
2	Subsection 9B.2.1.10.3	Modified Heading. Added details and paragraph summarizing plots presented in Appendix 9B.E.
2	Subsection 9B.2.1.10.4	Added reference to containment criteria in Chapter 3, Subsection 3.5.6. Removed reference to Open Item.
2	Subsection 9B.2.2.3	Clarified how containment internal structures are supported. Added detail on bioshield separation from RPV pedestal, and reference to Appendix 9B.G.
2	Subsection 9B.2.2.4	Added reference to high density concrete.
2	Subsection 9B.2.2.5	Edited to streamline text in reference to 9B.2.2.3.
2	Subsection 9B.2.2.6	Added details and reference to Appendix 9B.A.
2	Subsection 9B.2.2.10.2	Added detail in reference to Appendix 9B.A and Appendix 9B.F. Removed reference to Open Item.
2	Subsection 9B.2.2.10.3	Added detail to summarize plots in Appendix 9B.F.
2	Subsection 9B.2.3.3	Edited to reflect steel beam and RB exterior wall terminology.
2	Subsection 9B.2.3.4.1	Modified heading for Subsection 9B.2.3.4.1.
2	Subsection 9B.2.3.4.2	Added reference to Type B (studs).
2	Subsection 9B.2.3.4.4	Added reference to A325 Type 1.
2	Subsection 9B.2.3.4.6	Added reference to Clause 6.9.2 of CSA N291.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
2	Subsection 9B.2.3.5	Edited to reflect interaction of RB with adjacent structures consistent with Chapter 3.
2	Subsection 9B.2.3.6	Added details in Subsection 9B.2.3.6.
2	Subsection 9B.2.3.7	Edited to include reference to design of RB structure
2	Subsection 9B.2.3.9	Added detail on design of the SCCV, PRV pedestal and the bioshield.
2	Subsection 9B.2.3.10.2	Added reference to Appendices 9B.A, 9B.D and 9B.G. Removed reference to Open Item.
2	Subsection 9B.2.3.10.3	Modified heading. Added paragraphs and details in reference to Appendix 9B.G.
2	Subsection 9B.2.3.10.4	Edited reference to Security Annex for details on malevolent acts. Added reference to Chapter 3, Subsection 3.5.6 and details on RB robustness. Removed reference to Open Item.
2	Subsection 9B.3.1	Added detail on separation of RWB from other buildings and fire resistance rating
2	Subsection 9B.3.1.1	Edited to include reference to Chapter 2, Subsection 2.7.4.7.6.
2	Subsection 9B.3.2	Added detail on separation of CB from other buildings and fire resistance rating.
2	Subsection 9B.3.2.1	Edited detail in reference to Subsection 9B.3.1.1; included reference to Chapter 2, Subsection 2.7.4.7.6.
2	Subsection 9B.3.3	Added detail on separation of TB from other buildings and fire resistance rating.
2	Subsection 9B.3.4	Edited for clarity on DBE and design basis tornado loads; added detail on fire resistance rating.
2	Subsection 9B.3.5	Edited to refer to DNNP regulatory commitment.
2	Subsection 9B.3.5.2.1	Edited to streamline text in reference to Chapter 2.
2	Subsection 9B.4	Modified list and added references from appendices in Section 9B.4.
2	Table 9B-1 to Table 9B-4	Tables modified to include civil updates.
2	Figure 9B-1	Figure updated to provide clarity.
2	Figure 9B-2	Annotations added to figure to identify the inner and outer mat foundations.



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
2	Appendix 9B.A	All content in Appendix 9B.A, including tables and figures.
2	Appendix 9B.B	All content in Appendix 9B.B, including tables and figures.
2	Appendix 9B.C	All content in Appendix 9B.C, including tables.
2	Appendix 9B.D	All content in Appendix 9B.D, including tables and figures.
2	Appendix 9B.E	All content in Appendix 9B.E, including table and figures.
2	Appendix 9B.F	All content in Appendix 9B.F, including table and figures.
2	Appendix 9B.G	All content in Appendix 9B.G, including table and figures.

### ACRONYM LIST

Acronym	Explanation
3-D	3-Dimensional
AISC	American Institute of Steel Construction
ALARA	As-Low-As Reasonably Achievable
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
ATF	Acceleration Transfer Function
ATH	Acceleration Time History
BDBA	Beyond Design Basis Accident
BE	Best Estimate
BPVC	Boiler and Pressure Vessel Code
CEPSS	Containment Equipment and Piping Support Structure
CLE	Checking Level Earthquake
CRDH	Control Rod Drive Housing
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EQ	Environmental Qualification
FE	Finite Element
FIA	Foundation Interface Analysis
FIRS	Foundation Input Response Spectra
HVAC	Heating, Ventilation and Air-Conditioning
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
ISRS	In-Structure-Response Spectra
LB	Lower Bound
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
MCR	Main Control Room
NBC	National Building Code of Canada
NS-DBE	Non-Seismic Design Basis Earthquake
OBE	Operating Basis Earthquake
OGS	Offgas System
PBSRS	Performance Based Surface Response Spectra
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PLSA	Plant Services Area
POSAR	Preliminary Operating Safety Analysis Report
PSA	Probabilistic Safety Assessment
RB	Reactor Building
RPV	Reactor Pressure Vessel
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SIT	Structural Integrity Test
SMR	Small Modular Reactor
SRSS	Square-Root-of-the-Sum-of-Squares
SSC	Structures, Systems and Components
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
UB	Upper Bound

## TABLE OF CONTENTS

9B.0 CIVIL ENGINEERING WORKS AND STRUCTURES .....	1
9B.1 Foundations.....	2
9B.1.1 Structural Role.....	2
9B.1.2 Safety Design Bases .....	2
9B.1.3 Structural Description .....	2
9B.1.4 Materials.....	3
9B.1.5 Interfaces with Other Equipment or Systems .....	3
9B.1.6 System and Equipment Operation .....	3
9B.1.7 Instrumentation and Control.....	3
9B.1.8 Monitoring, Testing, Inspection and Maintenance .....	3
9B.1.9 Performance and Safety Evaluation.....	4
9B.2 Integrated Reactor Building .....	4
9B.2.1 Containment .....	4
9B.2.2 Containment Internal Structures .....	11
9B.2.3 Reactor Building .....	14
9B.3 Other Structures .....	21
9B.3.1 Radwaste Building.....	21
9B.3.2 Control Building .....	22
9B.3.3 Turbine Building.....	24
9B.3.4 Reactor Auxiliary Bay .....	25
9B.3.5 Pumphouse/Forebay and Tunnels .....	26
9B.3.6 Fire Pump Enclosure .....	27
9B.4 References.....	28
9B.5 Appendices.....	38
APPENDIX 9B.A – FINITE ELEMENT MODEL USED IN STRUCTURAL ANALYSES OF INTEGRATED REACTOR .....	39
9B.6 Introduction.....	39
9B.6.1 Integrated Reactor Building Finite Element Model .....	39
9B.6.2 Seismic SSI Analysis Model .....	41
9B.6.3 1-g Static SSI Analysis Model.....	43
9B.6.4 Static and Quasi-Static Analysis Models.....	44
9B.6.5 Thermal Stress Analysis Model.....	44
9B.6.6 Validation of FE Models.....	44

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

APPENDIX 9B.B – SEISMIC SOIL-STRUCTURE INTERACTION ANALYSIS .....	65
9B.7 Introduction.....	65
9B.7.1 SSI Analysis Methodology .....	65
9B.7.2 Key Nodal Locations.....	66
9B.7.3 Key Structural Members .....	66
9B.7.4 Design Basis Seismic Analysis Cases and Selected Frequencies of Analysis .....	66
9B.7.5 DNNP Seismic Response Characteristic .....	67
9B.7.6 Sensitivity Analysis Cases .....	69
9B.7.7 Bounding DBE Structural Stress Demands.....	73
9B.7.8 Bounding DBE In-Structure Demands.....	74
9B.7.9 Maximum DBE Displacements.....	74
APPENDIX 9B.C – INTERACTION EVALUATIONS RESULTS .....	119
9B.8 Introduction.....	119
9B.8.1 Seismic Interaction Evaluations .....	119
APPENDIX 9B.D – RESPONSES TO STATIC LOADS.....	125
9B.9 Introduction.....	125
9B.9.1 Design Parameters Considered in the Structural Analyses .....	125
9B.9.2 1-g Static SSI Analysis .....	127
9B.9.3 Static and Quasi-Static Analysis .....	128
9B.9.4 Thermal Stress Analysis .....	130
9B.9.5 Conclusions .....	131
APPENDIX 9B.E – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT .....	147
9B.10 Introduction.....	147
9B.10.1 SCCV Design Methodology .....	147
9B.10.2 Bounding Load Combinations and Design Demands .....	147
9B.10.3 Structural Capacity of SCCV Steel Bricks™ .....	147
9B.10.4 Structural Design Evaluation.....	148
9B.10.5 Critical SCCV Steel Brick™ Connections .....	149
APPENDIX 9B.F – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT INTERNAL STRUCTURES .....	155
9B.11 Introduction.....	155
9B.11.1 Pedestal Design Methodology .....	155
9B.11.2 Bounding Load Combinations and Design Demands .....	155

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

9B.11.3 Pedestal Steel Bricks™ Capacity .....	155
9B.11.4 Structural Design Evaluation.....	155
APPENDIX 9B.G – DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE .....	158
9B.12Introduction.....	158
9B.12.1 Design Methodology for the Reactor Building .....	158
9B.12.2 Bounding Load Combinations and Design Demands .....	158
9B.12.3 RB Steel Bricks™ Capacity .....	158
9B.12.4 Structural Design Evaluation.....	158
9B.12.5 Critical Steel Brick™ Connections in the Reactor Building .....	159

**LIST OF TABLES**

Table 9B-1: Load Combinations and Load Factors for SCCV.....	31
Table 9B-2: Load Combinations and Load Factors for Containment Closure Head.....	32
Table 9B-3: Load Combinations and Load Factors for Other Class MC Components .....	33
Table 9B-4: Load Combinations, Load Factors and Acceptance Criteria for RB Safety Class Steel Bricks™ Modules and Steel Structures .....	34
Table 9B.6-1: Integrated RB Analyses Cases and Model Requirements.....	47
Table 9B.6-2: Validation of RPV Lumped Mass Stick Model .....	49
Table 9B.7-1: Key Nodal Locations.....	75
Table 9B.7-2: Key Structural Members .....	75
Table 9B.7-3: Dominant Frequencies for SSI System Response .....	75
Table 9B.7-4: Effect of Subgrade Variations on Seismic Displacement Responses .....	76
Table 9B.7-5: Maximum Displacement at Key Nodal Locations Relative to Free-field.....	76
Table 9B.8-1: Subgrade Variation Effects on Seismic Driving Forces .....	122
Table 9B.8-2: DBE Seismic Forces at the Base of Power Block Structures.....	122
Table 9B.8-3: DBE Effective Acceleration Demands at the Base of Power Block Structures...	122
Table 9B.8-4: DBE Sliding Stability for Power Block Structures .....	123
Table 9B.8-5: DBE Overturning Stability for Power Block Structures.....	123
Table 9B.8-6: DBE Seismic Bearing Stress at the Base of Power Block Structures .....	123
Table 9B.8-7: Maximum Relative Displacements between the Reactor Building and Turbine Building.....	124
Table 9B.8-8: DBE Maximum Relative Displacements between the Reactor Building and Surrounding Power Block Structures .....	124
Table 9B.9-1: Equivalent Linear Static Subgrade Profiles .....	133
Table 9B.9-2: Bounding Tornado Design Parameters .....	133
Table 9B.9-3: Tornado Missile Spectrum and Maximum Horizontal Speed .....	133
Table 9B.9-4: Containment Pressure Loading.....	134
Table 9B.9-5: Summer/Winter Thermal Conditions in Reactor Building and SCCV During Normal Operations.....	134
Table 9B.9-6: Summer/Winter Thermal Conditions in Reactor Building and SCCV During DBA .....	134
Table 9B.9-7: Significant Static and Quasi-Static Load Cases .....	135
Table 9B.9-8: Significant Thermal Load Cases .....	136
Table 9B.10-1: Design Parameters for the SCCV Steel Bricks™.....	150
Table 9B.11-1: Design Parameters for Steel Bricks™ in the Reactor Building.....	156
Table 9B.12-1: Design Parameters for Steel Bricks™ in the Reactor Building.....	161

## LIST OF FIGURES

Figure 9B-1: BWRX-300 Integrated RB Structure Including Common Mat Foundation.....	35
Figure 9B-2: Plan View of BWRX-300 Common Mat Foundation .....	36
Figure 9B-3: BWRX-300 Containment Internal Structural Steel.....	37
Figure 9B.6-1: Overview of Integrated Reactor Building Finite Element Model.....	50
Figure 9B.6-2: Steel-Plate Composite Containment Vessel Structural Model with Penetrations	51
Figure 9B.6-3: Typical Steel Bricks™ Section .....	52
Figure 9B.6-4: Semi-Rigid Connection Locations .....	53
Figure 9B.6-5: Integrated Reactor Building Seismic SSI Analysis Model (Excluding Free-Field Subgrade Model) .....	54
Figure 9B.6-6: Turbine Generator Pedestal and Isolated Foundation at Grade .....	55
Figure 9B.6-7: Excavated Subgrade Volume Element Model .....	55
Figure 9B.6-8: Finite Element Model for Calculation of Subgrade Impedance .....	56
Figure 9B.6-9: Super-Element and Reactor Building Contact Springs.....	57
Figure 9B.6-10: Dynamic Comparison at Centre of Reactor Building Roof in Three Orthogonal Directions.....	58
Figure 9B.6-11: SCCV Top Slab Comparison .....	59
Figure 9B.6-12: Cumulative Mass Participation Comparison for the Radwaste Building Refined and Coarse Models.....	60
Figure 9B.6-13: Cumulative Mass Participation Comparison for the Control Building Refined and Coarse Models.....	61
Figure 9B.6-14: Cumulative Mass Participation Comparison for the Turbine Building Steel Shell Refined and Coarse Models.....	62
Figure 9B.6-15: Cumulative Mass Participation Comparison for the Shield Wall Area Structure in the Turbine Building .....	63
Figure 9B.6-16: Cumulative Mass Participation Comparison for the Reactor Auxiliary Bay Refined and Coarse Models.....	64
Figure 9B.7-1: Averaging of In-Structure Response Spectra from Five Sets of Acceleration Time History Results.....	77
Figure 9B.7-2: Enveloping and Broadening of In-Structure Response Spectra .....	77
Figure 9B.7-3: Key Nodal Locations and Structural Members .....	78
Figure 9B.7-4: Interpolation of Acceleration Transfer Functions .....	79
Figure 9B.7-5: Effects of Subgrade Variation on the Normalized Acceleration Time Histories...	80
Figure 9B.7-6: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – LB Case.....	81
Figure 9B.7-7: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – BE Case .....	81



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 9B.7-8: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – UB Case .....	81
Figure 9B.7-9: Membrane Vertical Forces – Reactor Building Exterior Wall – LB Case.....	82
Figure 9B.7-10: Membrane Vertical Forces – Reactor Building Exterior Wall – BE Case .....	82
Figure 9B.7-11: Membrane Vertical Forces – Reactor Building Exterior Wall – UB Case .....	82
Figure 9B.7-12: In-Plane Shear Forces – Reactor Building Exterior Wall – LB Case.....	83
Figure 9B.7-13: In-Plane Shear Forces – Reactor Building Exterior Wall – BE Case .....	83
Figure 9B.7-14: In-Plane Shear Forces – Reactor Building Exterior Wall – UB Case .....	83
Figure 9B.7-15: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – LB Case.....	84
Figure 9B.7-16: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – BE Case .....	84
Figure 9B.7-17: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – UB Case .....	84
Figure 9B.7-18: In-Plane Shear Forces – Wing Walls – LB Case.....	85
Figure 9B.7-19: In-Plane Shear Forces – Wing Walls – BE Case .....	85
Figure 9B.7-20: In-Plane Shear Forces – Wing Walls – UB Case .....	85
Figure 9B.7-21: Effect of Subgrade Variations on In-Structure Response at Top of Pedestal....	86
Figure 9B.7-22: Effect of Subgrade Variations on In-Structure Response at Edge of Reactor Building Roof .....	87
Figure 9B.7-23: Structural Stiffness Variation Effects on Vertical Forces in the Reactor Building Exterior Wall.....	88
Figure 9B.7-24: Structural Stiffness Variation Effects on the In-Plane Shear in the Reactor Building Exterior Wall.....	88
Figure 9B.7-25: Structural Stiffness Variation Effects on SCCV Top Slab Bending Moment.....	89
Figure 9B.7-26: Structural Stiffness Variation Effects on the RPV Pedestal Vertical Force.....	90
Figure 9B.7-27: Structural Stiffness Variation Effects on the RPV Pedestal In-Plane Shear.....	90
Figure 9B.7-28: Structural Stiffness Variation Effects on In-Structure Response Spectra for Operating Floor Slab.....	91
Figure 9B.7-29: Comparison of Dynamic and Static Lateral Earth Pressures .....	92
Figure 9B.7-30: Soil Separation Effects on Vertical Forces in the Reactor Building Exterior Wall .....	93
Figure 9B.7-31: Soil Separation Effects on the In-Plane Shear Force in the Reactor Building Exterior Wall .....	93
Figure 9B.7-32: Soil Separation Effects on SCCV Top Slab Bending Moment .....	94
Figure 9B.7-33: Soil Separation Effects on RPV Pedestal In-Plane Shear .....	94
Figure 9B.7-34: Soil Separation Effects on In-Structure Response Spectra for the Reactor Building Exterior Wall at Grade .....	95

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 9B.7-35: Soil Separation Effects on In-Structure Response Spectra for Top of Pedestal .....	96
Figure 9B.7-36: Saturated and Unsaturated BE Compression Wave Velocity Profiles .....	97
Figure 9B.7-37: Groundwater Variation Effects on Tangential (Hoop) Forces on the Reactor Building Exterior Wall.....	98
Figure 9B.7-38: Groundwater Variation Effects on In-Plane Shear in the Reactor Building Exterior Wall .....	98
Figure 9B.7-39: Groundwater Variation Effects on SCCV Top Slab Bending Moment.....	99
Figure 9B.7-40: Groundwater Variation Effects on RPV Pedestal In-Plane Shear.....	99
Figure 9B.7-41: Groundwater Variation Effects on In-Structure Response Spectra for Reactor Building Exterior Wall at Grade.....	100
Figure 9B.7-42: Groundwater Variation Effects on In-Structure Response Spectra for Operating Floor Slab.....	101
Figure 9B.7-43: Excavation Support and Backfill Effects on Vertical Forces in the RB Exterior Wall .....	102
Figure 9B.7-44: Excavation Support and Backfill Effects on In-Plane Shear in the RB Exterior Wall .....	102
Figure 9B.7-45: Excavation Support and Backfill Effects on RPV Pedestal In-Plane Shear ....	103
Figure 9B.7-46: Excavation Support and Backfill Effects on SCCV Top Slab Bending Moment .....	103
Figure 9B.7-47: Effects of Excavation Support and Backfill on In-Structure Response Spectra for Operating Floor Slab .....	104
Figure 9B.7-48: Effects of Enhanced Input Motions on In-Structure Response Spectra for Edge of RB Roof.....	105
Figure 9B.7-49: SSI Parameter Effects on In-Structure Response Spectra for RB Exterior Wall at Grade.....	106
Figure 9B.7-50: Seismic Design Basis Moment Demands for SCCV Top Slab.....	107
Figure 9B.7-51: Seismic Design Basis Membrane Forces for SCCV Wall .....	108
Figure 9B.7-52: Seismic Design Basis Moment Demands for SCCV Wall.....	109
Figure 9B.7-53: Seismic Design Basis Membrane Forces for RPV Pedestal.....	110
Figure 9B.7-54: Seismic Design Basis Membrane Forces for Reactor Building Exterior Wall..	111
Figure 9B.7-55: Seismic Design Basis Moment Demands for Reactor Building Exterior Wall..	112
Figure 9B.7-56: Seismic Design Basis Membrane Forces for Reactor Building Wing Walls ....	113
Figure 9B.7-57: Bounding Seismic In-Structure Response Spectra at Centre of Common Mat Foundation.....	114
Figure 9B.7-58: Bounding Seismic In-Structure Response Spectra at Top of RPV Pedestal...	115
Figure 9B.7-59: Bounding Seismic In-Structure Response Spectra at Edge of Common Mat Foundation.....	116

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 9B.7-60: Bounding Seismic In-Structure Response Spectra for the Operating Floor Slab .....	117
Figure 9B.7-61: Bounding Seismic In-Structure Response Spectra at Centre of RB Roof .....	118
Figure 9B.9-1: Compression Hoop Forces on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads .....	136
Figure 9B.9-2: Vertical Compression Forces on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads .....	137
Figure 9B.9-3: Horizontal Bending Moments on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads .....	137
Figure 9B.9-4: Out-of-Plane Shear on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads .....	138
Figure 9B.9-5: Compression Hoop Forces on the SCCV Wall from Unfactored Dead and Static Earth Pressure Loads .....	138
Figure 9B.9-6: Vertical Compression Forces on the SCCV Wall from Unfactored Dead and Static Earth Pressure Loads .....	139
Figure 9B.9-7: Pool Walls Deformed Shape due to Hydrostatic Loads .....	139
Figure 9B.9-8: Deformed Shape of Integrated Reactor Building due to Accidental Pressure ...	140
Figure 9B.9-9: Deformed Shape due to Groundwater Pressure on Reactor Building Exterior Wall .....	141
Figure 9B.9-10: Deformed Shape due to Additional Rock Pressure on Reactor Building Exterior Wall .....	142
Figure 9B.9-11: Deformed Shape of Integrated Reactor Building Under Normal Operating Winter Temperatures .....	143
Figure 9B.9-12: Deformed Shape of Integrated Reactor Building Under DBA temperatures during Winter .....	144
Figure 9B.9-13: In-Plane Forces due to DBA Temperatures During Winter .....	145
Figure 9B.9-14: Bending Moments due to DBA Temperatures During Winter .....	146
Figure 9B.10-1: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for SCCV Steel Plates Under Testing Load Category (SCCV-1) .....	150
Figure 9B.10-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the SCCV Top Slab Steel Plates Under Testing Load Category (SCCV-1) .....	151
Figure 9B.10-3: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the Inner Mat Foundation Concrete Infill Under Normal Load Category (SCCV-3) .....	151
Figure 9B.10-4: Demand-to-Capacity Ratios for the Inner Mat Foundation Under Abnormal Load Category (SCCV-7) .....	152
Figure 9B.10-5: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Wall Under Abnormal Load Category (SCCV-7) .....	153
Figure 9B.10-6: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Top Slab Under Abnormal Load Category (SCCV-7) .....	153

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Figure 9B.10-7: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Top Slab Under Abnormal/Extreme Environmental Load Category (SCCV-11).....	154
Figure 9B.10-8: Idealized SCCV Wall-to-Mat Foundation Connections .....	154
Figure 9B.11-1: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RPV Pedestal Under Normal Load Category (RB-2/3).....	156
Figure 9B.11-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RPV Pedestal Under Abnormal Load Category (RB-9).....	157
Figure 9B.12-1: Demand-to-Capacity Ratios for Out-of-Plane Shear in the Outer Mat Foundation Under Normal Load Category (RB-2/3) .....	161
Figure 9B.12-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior wall Below Grade Under Normal Load Category (RB-2/3) .....	162
Figure 9B.12-3: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Above Grade Under Normal Load Category (RB-2/3) .....	162
Figure 9B.12-4: Demand-to-Capacity Ratios for Out-of-Plane Shear in the Outer Mat Foundation Under Abnormal Load Category (RB-9) .....	163
Figure 9B.12-5: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Below Grade Under Abnormal Load Category (RB-9) .....	163
Figure 9B.12-6: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Above Grade Under Abnormal Load Category (RB-9) .....	164
Figure 9B.12-7: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for Wing Walls Under Abnormal Load Category (RB-9) .....	164
Figure 9B.12-8: Idealized Reactor Building Exterior Wall-to-Mat Foundation Connection.....	165
Figure 9B.12-9: Typical Reactor Building Wing Wall-to-Mat Foundation Connection.....	165
Figure 9B.12-10: Typical Wing Wall and Floor Connections to Reactor Building and SCCV Walls.....	166

## **9B.0 CIVIL ENGINEERING WORKS AND STRUCTURES**

Chapter 9B describes how the general design requirements specified in Chapter 3, Section 3.5 for the Seismic Category A integrated Reactor Building (RB) are complied with in the design of the structure, with Section 9B.1 focusing on the common mat foundation supporting the integrated RB and Section 9B.2 focusing on each of the RB integrated structures. As mentioned in Chapter 3, Section 3.5, reference to the integrated RB is inclusive of the RB, containment, and containment internals, whereas RB is used to refer to the RB structure outside of containment.

Also presented in Chapter 9B are the general design requirements for the Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), Reactor Auxiliary Bay, Pumphouse/Forebay structures and tunnels, and Fire Pump Enclosure. Information presented for these structures in this chapter is commensurate with their importance to safety. Remaining plant structures shown in Chapter 1, Appendix A, Figure A1.4-1 are not covered since they are not credited in the safety analysis.

Below are the key PSAR chapters that should be reviewed along with this chapter:

1. Chapter 1 which provides the Darlington New Nuclear Project (DNNP) general site and facility layout, a description of the BWRX-300 buildings, plant operational modes, principles of safety management and applicable codes and standards utilized in the design.
2. Chapter 2 which describes the characteristics of the DNNP site on which the BWRX-300 facility is constructed.
3. Chapter 3, Section 3.1, which provides the general design aspects and Defence-in-Depth safety framework utilized in the BWRX-300 design.
4. Chapter 3, Section 3.2, which provides the general classification of BWRX-300 Structures, Systems and Components (SSC) and the approach used to establish these classifications.
5. Chapter 3, Sections 3.3 and 3.4, which provide methodology and general design requirements for protection against the effects of external and internal hazards.
6. Chapter 3, Section 3.5, which provides general design aspects including methodology and codes and standards used in the design of seismic category buildings and civil engineering structures.
7. Chapter 6 which describes the containment and its associated systems, engineered safety features included to mitigate the consequences of anticipated operational occurrences or postulated design basis accidents, and habitability systems.
8. Chapter 9A which discusses BWRX-300 auxiliary systems including the Heating, Ventilation and Air-Conditioning (HVAC), the Fire Protection systems, overhead lifting equipment and floor drain systems.
9. Chapter 12 which describes the administrative programs and procedures, in conjunction with facility design, that ensure occupational radiation exposure to personnel is kept As-Low-As Reasonably Achievable (ALARA).
10. Chapter 15 which discusses Design Extension Conditions (DECs) and Beyond Design Basis Accidents (BDBAs) considered in the design and documents safety analysis results used to confirm adequacy of design.
11. BWRX-300 Security Annex, which evaluates the ability of the integrated RB to withstand regulatory defined threats.

## **9B.1 Foundations**

This section demonstrates how the design of the common Steel Bricks™ mat foundation supporting the integrated RB structure complies with the design requirements in Chapter 3, Subsection 3.5.2. For the design of the adjacent Power Block foundations, refer to Section 9B.3.

### **9B.1.1 Structural Role**

The common Steel Bricks™ mat foundation supports the deeply embedded integrated RB structure and provides the mechanism for transferring all applicable loads from the integrated RB structure to the surrounding subgrade.

### **9B.1.2 Safety Design Bases**

The seismic category of the containment and RB common mat foundation is the same as that of the superstructures. In compliance with Section 7.15.1 of CNSC REGDOC-2.5.2 (Reference 9B.4-1), the common mat foundation is evaluated and designed to ensure the stability and structural integrity of the integrated RB under normal operation, Anticipated Operational Occurrences (AOO) and Design Basis Accidents (DBAs). The robustness of the common mat foundation design under DEC is also considered in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1 and is discussed in Subsection 9B.2.1.10.

Since the integrated RB is deeply embedded in soil and rock, extensive geotechnical investigations are performed following the guidelines in NEDO-33914 (Reference 9B.4-2) to ensure proper soil and rock parameters, and subgrade modeling assumptions, are established for the Soil-Structure Interaction (SSI) of the deeply embedded integrated RB including foundation. Bounding site-specific subgrade parameters used in the analysis and design of the integrated RB are presented in Chapter 3, Tables 3.5-1 and 3.5-2. Subgrade modeling assumptions for the SSI of the deeply embedded integrated RB are described in Chapter 3, Subsection 3.5.1.1.

Stability evaluation of the common mat foundation is performed using the criteria in Chapter 3, Subsection 3.5.2.2. Consistent with CNSC REGDOC-2.5.2, Section 7.15.1, a potential liquefiable evaluation of the soil strata is performed, as discussed in Chapter 2, Subsection 2.7.4.7.6, and the effects of fluctuating groundwater are considered, in the evaluation of soil capacity. This is done to preclude total and differential settlement of the common mat foundation that can challenge the functions of Seismic Category A or B SSC in the integrated RB.

### **9B.1.3 Structural Description**

The deeply embedded cylindrical integrated RB structure is founded on a common circular mat foundation that supports the containment, containment internal structures and RB. This common mat foundation is made of Steel Bricks™ modules, same as the supported superstructures, and is deeply embedded in the rock. As mentioned in Chapter 3, Subsection 3.5.2.1, the containment boundary for the common mat foundation extends to the outer perimeter of the Steel-plate Composite Containment Vessel (SCCV) as shown in Figure 9B-1. The mat foundation supporting the SCCV is referred to as the inner mat foundation (See Figure 9B-2). The mat foundation outside of the containment boundary is referred to as the outer mat foundation.

The walls of the RB and containment structures carry the vertical loads from the structures to the deeply embedded common mat foundation. Lateral loads are transferred to these walls and to wing walls by the roof, floor diaphragms and Containment Equipment and Piping Support Structure (CEPSS). The below grade RB exterior wall and mat foundation transmit, in turn, the loads to the surrounding subgrade. The transfer of the lateral loads is affected by the interaction of the deeply embedded integrated RB structure with the surrounding soil and rock materials as discussed in Chapter 3, Subsection 3.5.1.1.

Figure 9B-1, Figure 9B-2 and Table 1.5-2 in Chapter 1 provide dimensions, plan, and section views of the integrated RB common foundation.

#### **9B.1.4 Materials**

Refer to Sections 9B.2.1.4 and 9B.2.3.4 for a description of materials used for the common mat foundation supporting the BWRX-300 integrated RB. The selection of these materials takes into account the environmental effects in compliance with CNSC REGDOC-2.5.2, Section 7.15.1.

#### **9B.1.5 Interfaces with Other Equipment or Systems**

The behaviour of contact interfaces between the common circular mat foundation supporting the integrated RB structures and the surrounding supporting media is considered in the SSI analyses discussed in Chapter 3, Subsection 3.5.1.1. These analyses consider Structure-Soil-Structure Interaction (SSSI) due to the closeness of the RWB, CB, TB, and Reactor Auxiliary Bay to the RB as described in Chapter 3, Subsection 3.5.1.1.

To protect the mat foundation from damage following core melt, a corium shield (See Figure 9B-1) is installed at the lower portion of the SCCV, and Reactor Pressure Vessel (RPV) pedestal as described in Chapter 3, Subsection 3.5.6.1.

#### **9B.1.6 System and Equipment Operation**

Not applicable to structural design of the common mat foundation.

#### **9B.1.7 Instrumentation and Control**

As described in Chapter 3, Subsection 3.3.1.5, seismic instrumentation is installed inside the RB at the top of the common mat foundation in accordance with Clause 5.2.3 of CSA N289.5 (Reference 9B.4-3) to monitor the seismic motions that may occur for the lifecycle of the reactor facility, starting from commissioning to decommissioning.

Piezometers, settlement sensors and extensometers are also used as described in Subsection 9B.1.8 to monitor the magnitude and distribution of pore pressure and amount of deformation of the subgrade materials surrounding the integrated RB.

#### **9B.1.8 Monitoring, Testing, Inspection and Maintenance**

Complying with requirements of Section 7.14 in CNSC REGDOC-2.5.2, monitoring, testing, inspection, and maintenance programs are implemented in the BWRX-300 design to avoid settling and degradation of the common mat foundation.

As mentioned in Subsection 9B.1.7, field instrumentation is installed inside and outside of the RB shaft to measure the distribution of pore pressures around and below the RB shaft, total settlement and tilt of the RB shaft including foundation, and measurement of the surrounding structures following the recommendations in NEDO-33914, Section 3.4. The instrumentation provides recordings during excavation and continuing through the BWRX-300 plant operation that can be benchmarked against design estimates using results of Foundation Interface Analyses (FIA) as described in NEDO-33914, Section 4.3.4. Short-term and long-term settlement monitoring plans are also developed to detect both vertical and horizontal movements in and around the integrated RB, as well as differential distortion across the foundation footprint and differential settlements between the containment and RB portions of the common mat foundations.

Refer to Chapter 3, Subsections 3.5.5.5 and 3.5.5.6 for a description of the minimum requirements for construction, in-service testing and inspection procedures for the Steel Bricks™ structures.

### **9B.1.9 Performance and Safety Evaluation**

In order to address the specific requirements for the design of the BWRX-300 deeply embedded RB, subsurface site investigations are performed as described in Chapter 2, Section 2.7, following the guidelines of NEDO-33914, Section 3.1, considering the combined footprint areas of the integrated RB common foundation and the adjacent RWB, CB, TB and Reactor Auxiliary Bay foundations. The extended area considered by the BWRX-300 subsurface site investigation ensures an adequate characterization of the subsurface conditions under the RWB, CB, TB and Reactor Auxiliary Bay foundations and resulting surcharge loads, which are important for the design of the deeply embedded RB structure and seismic design of integrated RB Seismic Category A and B SSC.

Stability evaluations for the common mat foundation under external hazards specified in Chapter 3, Section 3.3 are performed using results from the FIA (See Chapter 2, Subsection 2.11.3). Allowable bearing capacities for the common mat foundation taking into account the dry and submerged conditions at the site are presented in Chapter 2, Subsection 2.7.5.1. Stability and bearing capacity evaluations are performed for the integrated RB discussed in Section 9B.2 to ensure that the factors of safety listed in Chapter 3, Table 3.5-3 are met.

The design of the mat foundation portion within the perimeter or exterior surface of the SCCV, i.e., within the containment boundary, meets the design requirements of NEDC-33926P (Reference 9B.4-4). The design of the remaining portion meets the design requirements of ANSI/AISC N690 (Reference 9B.4-5) and NEDC-33926P.

Suitable provisions are provided to ensure that the common mat foundation is watertight as discussed in Chapter 3, Subsection 3.3.3.

### **9B.2 Integrated Reactor Building**

The integrated RB structure consists of the containment, containment internal structures and RB. The containment structure presented in Subsection 9B.2.1 protects and acts as a leak-tight boundary for the containment and associated systems discussed in Chapter 6, Section 6.3. Internal structures housed in the containment are described in Subsection 9B.2.2 and include the pedestal that supports the RPV and structures supporting piping and equipment within the containment.

The containment structure is completely enclosed within the deeply embedded RB structure presented in Subsection 9B.2.3. The RB is designed to protect the containment structure from external hazards (e.g., wind loads, tornado loads, aircraft hazard, missiles) and external beyond design basis scenarios. Besides the connection at the common mat foundation, the RB and containment structures are also connected at the top slab and through the RB wing walls and floor slabs at intermediate floor levels. The containment top slab directly supports the reactor cavity pool and Isolation Condenser System (ICS) pools and extends beyond the containment wall boundary to the RB exterior wall providing support for the fuel pool.

#### **9B.2.1 Containment**

As mentioned in Chapter 3, Subsection 3.5.3, the BWRX-300 containment comprises the SCCV, a steel containment closure head and other Class MC components.

##### **9B.2.1.1 Structural Role**

The primary functions of the containment structure are:

1. To enclose and support the RPV and its connected piping systems



2. To act as a leak-tight pressure boundary confining radioactive substances in different plant states, including normal operation, AOO, DBAs, and DEC's
3. To provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOO, DBAs, and DEC's
4. To provide in conjunction with the surrounding RB, protection for plant safety equipment and systems against external hazards and human induced events described in Chapter 3, Section 3.3

#### **9B.2.1.2 Safety Design Bases**

Per Chapter 3, Table 3.3-1, the BWRX-300 containment is a Seismic Category A structure.

As mentioned in Section 9B.2.1.1, the main function of the containment structure is to enclose and support containment systems and piping, and to act as a leak-tight pressure boundary confining radioactive substances in the four plant states groups identified in CNSC REGDOC-2.5.2, Section 7.3 (normal operation, AOOs, DBAs, and DEC's) in compliance with CNSC REGDOC-2.5.2, Section 8.6.2.

To perform its intended safety functions, the containment structure is designed to resist potential internal flooding, overpressures, underpressures, temperatures and dynamic effects as a result of internal missile generation and high energy line breaks discussed in Chapter 3, Section 3.4, with sufficient margins of safety in compliance with requirements in Section 8.6.2 of CNSC REGDOC-2.5.2. Being enclosed within the RB, the containment is also protected from the external hazards discussed in Chapter 3, Section 3.3.

The design bases and the various modes of operation of safety features and systems used to ensure the confinement functions and leak tightness of the containment are described in Chapter 6, Section 6.3.

Design bases for the robustness of the containment against overpressure, combustible gas pressures, seismic, elevated temperature and malevolent acts are discussed in Chapter 3, Subsection 3.5.6, Chapter 6, Subsection 6.3.5, and in Subsection 9B.2.1.10.

Design bases for the RB enveloping the containment structure and protecting it from external hazards and external beyond design basis scenarios are discussed in Subsection 9B.2.3.

#### **9B.2.1.3 Structural Description**

The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The BWRX-300 containment is a vertical cylinder comprised of the SCCV, consisting of Steel Bricks™ cylindrical wall, mat foundation, and top slab, and a steel containment closure head (See Figure 9B-1). The containment structure is completely enclosed within the deeply embedded RB and includes personnel/equipment hatches, containment penetrations, and other safety components as shown in Figure 9B-1. The SCCV is also integrated with the RB and the integrated structure is supported by a common Steel Bricks™ mat foundation.

The containment closure head that covers the opening in the SCCV top slab over the RPV is a removable steel torispherical head anchored to the SCCV top slab using a pair of mating flanges. It is designed for removal during reactor refueling and for replacement prior to reactor operation using the RB polar crane. Provisions are made for testing the flange seals to ensure the containment closure head can perform its pressure retaining functions. Stainless steel cladding is fixed to the outer surface of the containment closure head to protect it from the water in the reactor cavity pool above during normal operation and the clad thickness is determined in

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

accordance with ASME Boiler and Pressure Vessel Code (BPVC) Section III, Division 1, Subsection NE-3122 (Reference 9B.4-6).

As shown in Figure 9B-1, two access hatches are provided in the containment structure, at Level -8.5 m and Level -29.0 m, to provide access to the upper and lower portions of the containment. These access hatches serve the dual purpose of allowing personnel access to the containment and for movement of equipment. Each access hatch has two pressure-seated doors interlocked to prevent simultaneous opening of both doors and to ensure that one door is completely closed before the opposite door can be opened. The door operation is designed and constructed so either door may be operated from inside the containment, inside the lock, or from outside the containment. Also, each access hatch has a barrel, which is an inner round portion that is slid into the fixed portion of the sleeve that is embedded in the SCCV wall. The barrel of the access hatch can be removed if large equipment needs to be transported during outages.

The number of penetrations in the containment structure is kept to a minimum and the containment penetrations are designed for loading conditions applicable for the containment structure and to allow for periodic inspection and testing in compliance with Section 8.6.5 of CNSC REGDOC-2.5.2. Figure 9B-1 depicts some of these containment penetrations. The major piping penetrations through the SCCV are associated with main steam and feedwater lines. Piping and electrical penetrations into the SCCV are categorized into two basic types, depending on whether the penetration is subjected to a hot or cold operational environment. Cold penetrations pass through the SCCV wall and are embedded directly in it. Hot penetrations do not come in direct contact with the SCCV wall but are provided with a thermal sleeve, which is attached to the SCCV wall. The thermal sleeve is attached to the process pipe at distance from the SCCV wall to minimize conductive heat transfer to the SCCV wall. The sleeve length for hot penetrations is designed to meet the concrete temperature limitations specified in NEDC-33926P. SCCV electrical penetrations are sealed to the interior side of the SCCV pressure boundary. For more information on containment penetrations, refer to Chapter 6, Subsections 6.3.2.2 and 6.3.4.

#### **9B.2.1.4 Materials**

The SCCV is constructed using Steel Bricks™ that are advanced steel-plate composite elements comprised of inner and outer steel faceplates, a continuous diaphragm plate through the thickness of section (resulting from the bending and welding of steel faceplates), shear connectors, and the fill concrete. The inner and outer faceplates maintain their structural integrity to perform the load resisting function while the inner steel faceplates of the SCCV also perform a liner function to provide a leak-tight boundary.

The containment closure head and other Class MC components are made of steel. Stainless steel cladding is fixed to the outer surface of the containment closure head to protect it from the water in the reactor cavity pool above as discussed in Subsection 9B.2.1.3.

Complying with Sections 7.15.1 and 8.6.11 of CNSC REGDOC-2.5.2, the selection of containment design materials takes into account service temperatures and other conditions of the containment pressure boundary during operation, maintenance, testing, and postulated accident conditions to ensure fulfillment of its functions. Uncertainties in determining material properties, residual, steady-state and transient stresses, and the size of flaws are also considered in the design material selection. For more information on the coverings, coatings and materials used for the containment engineered safety features, refer to Chapter 6, Section 6.1.

The impact testing requirements for SCCV steel faceplates are in accordance with NEDC-33926P. The impact testing requirements for containment closure head and Class MC components steel materials is in accordance with ASME Section III Subsection NE-2300 for Class MC components.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Corrosion protective coatings and systems are applied on the exposed surface of the SCCV steel faceplates and interior surface of the containment closure head and Class MC components to protect the metal from corrosion and facilitate decontamination. Guidelines in U.S. NRC RG 1.54 (Reference 9B.4-7) and ASTM D5144 (Reference 9B.4-8) are used, as applicable, for application, maintenance, and qualification of permanent corrosion protective coatings.

For fire protection measures considered for the containment structure, refer to Subsection 9B.2.1.10.

Design specifications covering materials are in sufficient detail to ensure the structural design requirements are met. Properties of materials used in the construction of the containment are provided in the following subsections.

**9B.2.1.4.1 Steel Bricks™ Concrete Infill**

Design compressive strength for the concrete used for the construction of the SCCV is 34.5 MPa at 28 days.

For those areas of SCCV, portions of mat foundation within the SCCV, and other adjacent structural features subjected to elevated temperatures for long durations of time, the aggregates in concrete do not contain pyretic materials.

All concrete materials are approved prior to start of construction on the basis of their characteristics in test comparisons using applicable ASTM standard methods. Concrete aggregates and cement, conforming to requirements in material specifications, are obtained from approved sources. All structural concrete is batched and placed in accordance with NEDC-33926P.

Supplementary concrete sampling requirements are used as described in Section 3.2.1 of NEDO-33914 to address the effect of small volume of concrete placed for the BWRX-300.

**9B.2.1.4.2 Containment Steel Bricks™ Steel**

Steel Bricks™ steel components conform to the following:

- Steel faceplates/diaphragm plate: ASME SA-738 (Reference 9B.4-9), Grade B
- Studs: ASTM A108, Type B (Reference 9B.4-12) as permitted by ASME Section III, Division 2, Subsection CC (Reference 9B.4-13), MANDATORY APPENDIX D2-1, Table D2-1-2.2

**9B.2.1.4.3 Structural Steel and Appurtenances for the SCCV**

The structural steel and other appurtenances for the SCCV other than the primary Steel Bricks™ steel components covered above conform to the following:

- Structural Steel embedded plate and attachments: ASTM A572 (Reference 9B.4-14) Gr. 50 per ASME Code Case N-632 (Reference 9B.4-15)
- Penetrations: ASME SA-516 (Reference 9B.4-16) Gr. 60 or Gr. 70 or SA537 (Reference 9B.4-17) Class 2

**9B.2.1.4.4 Structural Steel and Appurtenances for the Containment Closure Head and Other Class MC Components**

Structural steel and other appurtenances for the containment closure head and other Class MC components conform to the following:

- Structural Steel: ASME SA-516 Gr. 60 or Gr. 70, SA537 Class 1 or Class 2 or SA-738 Gr. B

- Attachments: ASTM A572 Gr. 50
- Stainless Steel Cladding: ASME SA-240 (Reference 9B.4-18) Type 304L; cladding in accordance with ASME SA-264 (Reference 9B.4-19). Maximum carbon content of Type 304L  $\leq 0.020$  percent
- Bolting: ASME SA-193 (Reference 9B.4-20), Gr. B8, SA-437 (Reference 9B.4-21) Gr. B4B

#### **9B.2.1.5 Interfaces with Other Equipment or Systems**

The BWRX-300 containment is completely enclosed within the RB, with both structures resting on the common mat foundation that also supports the containment internal structures. As shown in Figure 9B-1 and Figure 9B-2, the containment is integrated with the RB structure at the common mat foundation, at the top slab, at the connections of wing walls and at intermediate floor levels. In addition to interfacing with the RPV pedestal at the mat foundation, the containment also interfaces with the containment internal steel structures at various elevations (See Figure 9B-1) and with the main steam and feedwater piping and other electrical and piping lines penetrating through the containment wall. Chapter 6, Subsection 6.3.2.2 provides more information on penetrations and piping into the containment structure.

The integrated RB model used in the SSI analyses discussed in Chapter 3, Subsection 3.5.1.1 appropriately incorporates the interfaces between the containment structure, containment internal structures, and RB to represent the vertical and lateral load transfer between these different structures for applicable internal and external loads as requested by CNSC REGDOC-1.1.2 (Reference 9B.4-22), Section 4.5.5.

The connections of the RB walls and floors to the outside face of the SCCV wall are outside ASME code jurisdiction, with the exception of attachment welds. Attachment welds are designed to follow ASME quality assurance and welding procedures, and inspection requirements.

#### **9B.2.1.6 System and Equipment Operation**

Refer to Chapter 6, Section 6.3 for a discussion of containment systems designed to ensure its safety functions, including confinement, under accident conditions.

#### **9B.2.1.7 Instrumentation and Control**

In accordance with Clause 5.2.3 of CSA N289.5, seismic instrumentation is installed close to the top of containment as described in Chapter 3, Subsection 3.3.1.5 to monitor the seismic motions during the lifecycle of the reactor facility.

Refer to Chapter 6, Section 6.3, and Chapter 7 for a discussion of systems and instrumentations used to monitor pressure inside the containment.

#### **9B.2.1.8 Monitoring, Testing, Inspection and Maintenance**

Containment safety functions are ensured throughout the life of the plant through monitoring, testing, inspections, and maintenance in compliance with CNSC REGDOC-2.5.2, Sections 7.14 and 8.6.3.

Testing and In-service inspection of the containment structure are performed per the criteria in Chapter 3, Subsection 3.5.3.6.

As stated in Chapter 3, Subsection 3.5.5.6, a research and development program is established to demonstrate that the design, testing and inspection methods of the Steel Bricks™ containment meet all applicable safety design requirements in compliance with CNSC REGDOC-2.5.2, Section 5.4.

The Structural Integrity Test (SIT)/Pre-Operational Proof Test pressure a) is at least 1.15 times the positive design pressure and b) is the negative design pressure, where applicable in compliance with NEDC-33926P.

Instrumentation provided to evaluate behaviour of the structure and the actual stress values during Pre-Operational Proof Test is also used to monitor the deformation of the containment structure at the time of the Integrated Leak Rate Test (ILRT) to ensure the elastic behaviour of the containment structure. The ILRT is performed as described in Chapter 6, Subsection 6.3.7.

Refer to Chapter 6, Section 6.3 and Chapter 7 for a discussion of engineered safety features, systems and instrumentation used to maintain the containment barrier for radiation control and protection.

#### **9B.2.1.9 Radiological Aspects**

The SCCV is designed to withstand the highest design basis accident pressure. The BWRX-300 SCCV is provided with overpressure protection by means of a containment inerting system as discussed in Chapter 9A, Subsection 9A.4.2. The design basis accident pressure for containment is evaluated in Chapter 15, Section 15.5 and the containment design pressure is discussed in Chapter 6, Subsection 6.3.2.

Containment response to radiological events such as Loss-of-Coolant Accident (LOCA) is discussed in Chapter 15, Subsection 15.5.4.6.

As mentioned in Chapter 6, Subsection 6.3.2, permanent radiation shielding is provided within and outside of the containment to limit doses to ALARA, with the SCCV structural concrete functioning as radiation shielding external to the containment, along with bioshield and the RPV pedestal (see Subsection 9B.2.2.3 and Figure 9B-1). Radiation dose varies in the containment structure by locations and the SCCV minimum wall and top slab thickness and minimum steel faceplate thicknesses are selected to provide a proper radiation shielding. For containment shielding evaluations, refer to Chapter 12, Section 12.4. For other design features used for radiation protection, including the ALARA structures criteria observed in the BWRX-300 design, refer to Chapter 12, Section 12.3.

The lower portion of the SCCV and RPV pedestal is also provided with a corium shield to prevent contact between the molten core and the Steel Bricks™ faceplates and concrete and RPV pedestal to mitigate the consequences of severe accident conditions as described in Chapter 3, Subsection 3.5.6.1. For more details on this corium shield, refer to Chapter 15, Appendix 15.B.

#### **9B.2.1.10 Performance and Safety Evaluation**

The BWRX-300 containment structure, including isolation of containment penetrations, is of a fail-safe design such that confinement of radioactive material can be achieved for a minimum of seven (7) days under worst-case environmental conditions, without credit taken for mitigating operator action.

To meet its functional and performance requirements listed in Subsection 9B.2.1.2, the BWRX-300 containment structure is evaluated and designed per the requirements in Chapter 3, Subsection 3.5.3, and its robustness evaluated for the DECAs described in Chapter 3, Subsection 3.5.6. To meet safety margin requirements in CNSC REGDOC-1.1.2, Section 4.5.5 and CNSC REGDOC-2.5.2, Sections 7.5 and 7.7, the containment design pressure is selected to bound DBAs with sufficient margin. As shown in Chapter 15, Subsection 15.5.4.6, the containment peak pressure resulting from the most limiting design basis accident is at least 10% lower than the containment design pressure. Similarly, the containment design temperature bounds the accident peak values in Chapter 15, Subsection 15.5.4.6.

Being enclosed within the RB, the containment structure is protected against external hazards discussed in Chapter 3, Section 3.3. The containment structure is also designed for reaction forces stemming from the pipe movements, jet impact, pipe whip and other internal missiles described in Chapter 3, Subsections 3.4.3 and 3.4.4.

The SCCV is designed as a floodable volume to assure core coverage in response to an accident, as described in Chapter 6, Subsection 6.3.2 and Table 9B-1.

Complying with CNSC REGDOC-2.10.1 (Reference 9B.4-23), CSA N293 (Reference 9B.4-24) and CSA N293S1 (Reference 9B.4-25), containment walls are to be fire rated with steel plates exposed to fire as discussed in Chapter 3, Subsection 3.4.1.1. Doors or HVAC dampers isolating the containment are also designed to be fire rated with fire/smoke dampers compatible with the rating of the fire barrier. Containment atmosphere is nitrogen-inerted during normal operation to preclude the initiation or propagation of a fire. Portable detection equipment and fire watches used inside containment during maintenance outages when the space is not inerted are discussed in Chapter 9A, Subsection 9A.6.6.

#### **9B.2.1.10.1 Design Loads and Load Combinations for Containment Structure**

The containment structure is analysed and designed for all credible conditions of loading, including normal loads, pre-operational testing loads, loads during severe and extreme environmental conditions and loads during abnormal plant conditions in compliance with CNSC REGDOC-2.5.2, Section 7.15.1. Loads used in the analysis and design of the containment are described in Chapter 3, Subsection 3.5.3.2.

Load combinations considered in the design take into account the probability of occurrence, loading time history and sequence of loads, as applicable, meeting the requirements in CNSC REGDOC-2.5.2, Section 7.15.1. Table 9B-1 presents load combinations used in the design of the SCCV portion of the containment. Table 9B-2 and Table 9B-3 summarize the load combinations and associated load factors used in the design of the containment closure head and other Class MC components, respectively. As shown in Table 9B-2 and Table 9B-3, combined loadings are categorized according to Level A, C and D service limits as defined in Subsection NE-3113 of ASME BPVC, Section III, Division 1. Since the Operating Basis Earthquake (OBE) is set at one-third of the Design Basis Earthquake (DBE) for design purposes as described in Chapter 3, Subsection 3.3.1, OBE load combinations are not considered. For the design of Class MC components (excluding containment closure head), the load combination involving post-flooding condition is considered in accordance with U.S. NUREG-0800 (Reference 9B.4-26), SRP 3.8.2.

As discussed in Chapter 3, Subsection 3.5.3.2, the portion of the BWRX-300 containment closure head and other Class MC components backed by concrete are designed for the load combinations and associated load factors in accordance with NEDC-33926P, supplemented by RG 1.136 (Reference 9B.4-27), as summarized in Table 9B-1.

The design of the containment structure, including SCCV, containment closure head, and other Class MC components, also meets thebdba/dec requirements as discussed in Chapter 3, Subsection 3.5.6 to comply with CNSC REGDOC-2.5.2, Section 7.15.1.

#### **9B.2.1.10.2 Containment Analysis and Design**

The design of the containment structure is based on the membrane forces, shear forces, and bending moment demands from the loads and load combinations defined in Subsection 9B.2.1.10.1.

These demands are obtained from the set of analysis described in Chapter 3, Subsection 3.5.1.1 performed using the Finite Element (FE) model of the integrated RB structure developed following

the criteria in Chapter 3, Subsections 3.3.1.2 and 3.5.1.1 (See Appendix 9B.A for details of model). As mentioned in Chapter 3, Subsection 3.3.1.2, larger penetrations, and openings are also included in the model.

The state of stress and behaviour of the containment around these openings is determined by use of acceptable analytical techniques, including but not limited to, the use of refined local shell or solid FE models incorporating the imposed displacements calculated from the global integrated FE model.

Key demands used in the design of the SCCV structure are provided in Appendix 9B.E. Design evaluation of the containment closure head, and Class MC components of the containment will be reported in the Preliminary Operating Safety Analysis Report (POSAR) in support of the Licence to Operate application.

#### **9B.2.1.10.3 Structural Design Margins**

To meet the structural integrity requirement in CNSC REGDOC-2.5.2, Section 7.15.1, safety margins of the containment structure are ensured by satisfying the requirements and acceptance criteria discussed in Chapter 3, Subsection 3.5.3.4. Bounding demand-to-capacity ratios for key components of the SCCV are provided in Appendix 9B.E to demonstrate the adequacy of the design.

Per the plots presented in Appendix 9B.E, the design of the BWRX-300 SCCV at the DNNP site include safety margins of at least 30% for the testing conditions, 40% for normal operating conditions, and 20% for abnormal conditions. The most critical loading conditions for the SCCV are the loads associated with containment DBA conditions, primarily the accident temperature loading and to a lesser extent the accidental pressures. The DNNP site-specific bounding seismic loads have a less significant impact on the design of the SCCV.

#### **9B.2.1.10.4 Robustness**

The robustness of the containment structure is evaluated per criteria outlined in Chapter 3, Subsections 3.3.7.4 and 3.5.6. Being enclosed in the lower portion of the deeply embedded RB, the containment structure is protected against malevolent acts, such as a large airplane crash, as demonstrated in the BWRX-300 Security Annex.

Design evaluation results demonstrating the robustness of the containment design per criteria outlined in Chapter 3, Subsection 3.5.6 will be provided in the POSAR in support of the Licence to Operate application.

The containment is designed to maintain its role as a leak-tight barrier for DECAs with severe core damage for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage in compliance with CNSC REGDOC-2.5.2, Section 7.3.4.1. The Level 2 Probabilistic Safety Assessment (PSA) performed for the POSAR describes the severe accident sequences leading to core damage and demonstrates operators have sufficient time to bring the plant to safe shutdown because the severe accidents are slow progressing for the BWRX-300.

Other design features included to ensure robustness of the containment are discussed in Chapter 15, Appendix 15A and Appendix 15B.

### **9B.2.2 Containment Internal Structures**

The BWRX-300 containment internal structures include the RPV pedestal, the bioshield surrounding the RPV pedestal, and the containment internal structural steel which consists of the CEPSS and the support floors at Level -21 m and -29 m as shown in Figure 9B-1.

#### **9B.2.2.1 Structural Role**

The primary functions of the RPV pedestal are:

1. To provide structural support to SSC such as the RPV, RPV Skirt Support, CEPSS, Stabilizers, and miscellaneous platforms
2. In conjunction with the bioshield, to provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOO, DBAs, and DEC conditions

The primary functions of the bioshield are:

1. In conjunction with the RPV pedestal, to provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states, including normal operation, AOO, DBAs and DEC conditions
2. To provide structural support to miscellaneous platforms

The primary functions of the CEPSS are:

1. To provide supports for SC1, SC2 and Non-Safety Class (SCN) piping, RPV Stabilizers, SC1 RPV isolation valves including actuator, and other components
2. To serve as support for miscellaneous platforms and associated floor grating, ladders/stairs and other apparatus needed for personnel access and equipment inspection and maintenance

The primary functions of the support floors at Level -21 m and -29 m are:

1. To provide support for components, including fans and coolers, associated with the SC3 Component Cooling System at Level -21 m
2. To provide support for miscellaneous SCN components at Level -29 m

#### **9B.2.2.2 Safety Design Bases**

As indicated in Chapter 3, Table 3.3-1, the BWRX-300 containment internal structures are Seismic Category A structures designed to support SSC inside the containment. The RPV pedestal and bioshield are also designed to provide radiation shielding to limit radiation dose during normal operation, AOO, DBAs and DEC conditions.

#### **9B.2.2.3 Structural Description**

The RPV pedestal is the cylindrical-shaped Steel Bricks™ structure that structurally supports the RPV and is integrated into the common mat foundation as shown in Figure 9B-1.

The RPV pedestal is equipped with a structural steel RPV bracket that runs on the inside at the top of the RPV pedestal and where the RPV skirt is anchored using anchor bolts. The RPV pedestal also provides structural support for the RPV bottom stabilizers and the CEPSS. The part of the RPV pedestal surrounding the core region of RPV helps to attenuate the radiation emanating from the RPV. Openings are provided in the RPV pedestal to permit the routing of necessary piping to the RPV, to permit in-service inspection of the RPV and piping, and to ensure personnel access into the under-vessel region.

The bioshield is an independent cylindrical-shaped steel-plate composite structure that surrounds the RPV pedestal (See Figure 9B-1), with both structures integrated into the common mat foundation. The bioshield is separated from the RPV pedestal by a seismic gap. The primary function of the bioshield is to attenuate the radiation emanating from the RPV in conjunction with the RPV pedestal. The bioshield also provides support to the steel platforms at levels -21 m and



-29 m. Similar to the RPV pedestal, openings are provided in the bioshield to permit the routing of necessary piping to the RPV, to permit in-service inspection of the RPV and piping, and to ensure personnel access into the under-vessel region.

Rigid overstrength connections (See Appendix 9B.G, Subsection 9B.12.5) are used to connect the RPV pedestal and bioshield to the common mat foundation.

The CEPSS is supported primarily by the RPV pedestal as shown in Figure 9B-1. The CEPSS framing connects directly to the RPV pedestal and bypasses the bioshield so that the latter remains independent of the RPV pedestal. Shear lug support connections with the SCCV are used to allow free thermal expansion of the CEPSS. The CEPSS consists of various structural components such as beams and columns as shown in Figure 9B-3. Hollow structural steel sections are used for the CEPSS beams and columns that are welded together with moment continuity. Wide flange or hollow structural steel sections are used to support the steel platform floors at Level -8.5 m, -21 m and -29 m. The steel platform at Level -8.5 m is supported by the CEPSS. The steel platform beams at level -21 m and -29 m are supported by the bioshield and SCCV wall. RPV stabilizers installed at the top of the CEPSS provide horizontal bracing close to the top of the RPV and another set of stabilizers supported from the RPV pedestal provide support near the bottom of the RPV.

#### **9B.2.2.4 Materials**

The BWRX-300 RPV pedestal is a Steel Bricks™ structure, the bioshield is a steel-plate composite structure, while the CEPSS and the support floors at Level -21 m and -29 m are made of steel.

Refer to Subsection 9B.2.3.4 for properties of steel and concrete materials used in the construction of the BWRX-300 containment internal structures. The selection of these materials takes into account the environmental effects in compliance with CNSC REGDOC-2.5.2, Section 7.15.1.

High density concrete used for shielding purposes is designed in accordance with U.S. NRC RG 1.69 (Reference 9B.4-28).

#### **9B.2.2.5 Interfaces with Other Equipment or Systems**

The RPV pedestal interfaces with the RPV through the RPV skirt support and horizontal stabilizers.

The CEPSS interfaces with the RPV at the upper stabilizer locations. Containment internal structures also interface with the support piping, equipment and general access platforms and stairs needed for personnel access and equipment inspection and maintenance.

#### **9B.2.2.6 System and Equipment Operation**

Supplemental equipment loads have been added to the structural FE models as discussed in Appendix 9B.A, Subsection 9B.6.1.5 to support mechanical and electrical equipment.

#### **9B.2.2.7 Instrumentation and Control**

In accordance with Clause 5.2.3 of CSA N289.5, seismic instrumentation is installed on the containment internal structures as described in Chapter 3, Subsection 3.3.1.5 to monitor the seismic motions during the lifecycle of the reactor facility.

For instrumentation used to monitor containment pressure and temperature, refer to Chapter 6, Subsection 6.3.2.6.

#### **9B.2.2.8 Monitoring, Testing, Inspection and Maintenance**

As described in Chapter 3, Subsection 3.5.4.6, a formal program for monitoring, testing and inspection is not required for containment internal structures. For more details, refer to Chapter 3, Subsection 3.5.4.6.

#### **9B.2.2.9 Radiological Aspects**

Penetrations through the containment wall are shielded to reduce radiation streaming in line with the ALARA criteria discussed in Chapter 12, Subsection 12.3.6.

Corrosion protective coatings and systems are also applied on the exposed surface of the steel faceplates of the RPV pedestal to protect the metal from corrosion and facilitate decontamination.

#### **9B.2.2.10 Performance and Safety Evaluation**

##### ***9B.2.2.10.1 Loads and Load Combinations***

Complying with CNSC REGDOC-2.5.2, Section 7.15.1, the BWRX-300 containment internal structures are designed for all applicable load combinations, with due consideration given to the probability of concurrence, loading time history and sequence of loads, as applicable.

Load combinations and load factors for the containment internal structures are covered by the comprehensive set of load combinations listed in Subsection 9B.2.3.10 used for the design of the RB structure, considering the difference in applicable design loads for the containment internal structures compared to the RB.

##### ***9B.2.2.10.2 Analysis and Design Procedures***

As mentioned in Chapter 3, Subsection 3.5.4.3, the containment internal structures are analysed and designed using the integrated RB FE model and following the same modeling and structural analysis procedures as for the RB (See Appendix 9B.A for details of the model)

Internal structural steel supporting piping systems within the containment are designed to carry piping dynamic loads without buckling and while remaining elastic. Beams and columns supporting pipe whip restraints allow inelastic deformations due to pipe rupture loads. Seismic Category A and B SSC impacted by these inelastic beam deformations are evaluated to verify that their safety functions are not compromised.

Key demands used in the design of the RPV pedestal are summarized in Appendix 9B.F. Design evaluation of other containment internal structures will be reported in the POSAR in support of the Licence to Operate application.

##### ***9B.2.2.10.3 Structural Design Margins***

To meet the structural integrity requirement in CNSC REGDOC-2.5.2, Section 7.15.1, safety margins of the containment internal structures are ensured by satisfying the requirements in Chapter 3, Subsection 3.5.4.4. Bounding demand-to-capacity ratios are provided in Appendix 9B.F to demonstrate the adequacy of the design of the RPV pedestal.

As demonstrated by the plots in Appendix 9B.F, the design of the RPV pedestal includes safety margins of at least 20%. Seismic loads in combination with the other abnormal condition loads govern the design of the RPV pedestal.

#### **9B.2.3 Reactor Building**

##### **9B.2.3.1 Structural Role**

The primary functions of the RB are:

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

1. To house and structurally support the reactor vessel, the containment structure, the reactor support structure of the primary reactor system and fuel handling equipment, biological shielding, and associated equipment and structures.
2. To provide adequate space for the operation, maintenance, and removal of equipment housed within the containment structure during periodic maintenance.
3. To provide protection for safety equipment from environmental and natural hazards phenomena, such as floods, winds, tornadoes, and earthquakes.
4. To provide protection for safety equipment from external (e.g., explosions and missiles from nearby transportation/industry or aircraft impact) and internal (e.g., pipe break or heavy load drop) hazards.
5. To support habitability functions for Secondary Control Room (SCR) such as radiological shielding and toxic gas isolation and passive cooling for human occupancy in addition to Environmental Qualification (EQ) for equipment.

#### **9B.2.3.2 Safety Design Bases**

Per Chapter 3, Table 3.3-1, the BWRX-300 RB is a Seismic Category A structure.

As mentioned in Subsection 9B.2.1.1, the main function of the RB structure is to protect the reactor vessel, the containment and safety class systems and equipment from external hazards that might impair their safety functions during normal operation, AOOs, DBAs, and DECAs in compliance with CNSC REGDOC-2.5.2, Section 7.3.

External hazards considered in the design of the RB are described in Chapter 3, Section 3.3.

#### **9B.2.3.3 Structural Description**

The RB is a cylindrical building made of Steel Bricks™ floors and walls, with steel beams supporting the Steel Bricks™ roof modules.

The cylindrical-shaped, below grade portion of the RB structure encloses the containment and protects the RPV, reactor support and safety systems, and the majority of vital and non-vital power supplies and equipment. While the majority of the RPV and SCCV is located below grade, the top elevation of the RPV and SCCV top slab is above grade. In addition, the above grade portion of RB structure houses the refueling floor, refueling and fuel handling systems, fuel pool, and polar crane.

Chapter 1, Table 1.5-2 provides key dimensions of the RB while Figure 9B-1 presents an elevation sectional view of the structure. Figure 9B-1 depicts the separation of the RB structure code boundary from the containment pressure boundary discussed in Subsection 9B.2.1.3. RB structural boundary includes the following areas:

1. Steel Bricks™ mat foundation between the SCCV wall and the RB exterior wall that functions as a common foundation for the RB and SCCV structures
2. RB below grade (from the mat foundation) to the support for the polar crane, including the roof, composed of Steel Bricks™ modules, with structural steel beams
3. ICS pools, fuel pool, equipment pool, and the reactor cavity pool composed of Steel Bricks™ walls that are lined for corrosion protection with leakage detection system as applicable
4. Rooms at several elevation levels outside the containment, structurally integral with the containment structure

As shown in Figure 9B-1, the SCCV and the RB structure are integrated at the connections of wing walls and elevated slabs between the mat foundation and the reactor cavity pool top slab elevation of the structure. The structures share a common mat foundation as discussed in Section 9B.1.

#### **9B.2.3.4 Materials**

This section presents the properties of materials used in the construction of the RB. The selection of these materials takes into account the environmental effects described in Chapter 3, Sections 3.3 and 3.4 in compliance with CNSC REGDOC-2.5.2, Section 7.15.1.

The RB is designed to withstand intrusion by regulatory defined threats by selecting design parameters for the Steel Bricks™, such as steel thickness, steel ductility, and concrete thickness. The ability of Steel Bricks™ to resist malevolent acts is demonstrated in the BWRX-300 Security Annex.

##### **9B.2.3.4.1 Steel Bricks™ Concrete Infill**

Design compressive strength for the self-consolidating concrete used for the construction of the RB is 34.5 MPa at 28 days.

Concrete ingredients used in the RB Steel Bricks™ modules are tested, batched, and mixed in accordance with ACI 349 (Reference 9B.4-29).

##### **9B.2.3.4.2 Steel Bricks™ Steel**

Steel Bricks™ steel components conform to the following:

- Steel faceplates/diaphragm plate: ASTM A572 Gr. 50
- Studs: ASTM A108, Type B

##### **9B.2.3.4.3 Structural Steel**

Structural steel conforms to ASTM A572 Gr.50.

##### **9B.2.3.4.4 Steel Connections**

Bolts used in structural steel bolted connections are ASTM F3125 (Reference 9B.4-30), Grade A325 Type 1.

Bolts of the following grades may be used otherwise, when deemed necessary:

- ASTM F3125, Grade A490 Type 1 or A325 Type 1
- ASTM A354 (Reference 9B.4-31), Grade BD
- ASTM A449 (Reference 9B.4-32), Type 1
- ASTM A193, for high service temperature applications, grade is subject to specific design requirements.

For SC1 structures, ASTM A307 (Reference 9B.4-33) carbon steel fasteners are not to be used in structural connections.

##### **9B.2.3.4.5 Welding**

Welding materials for the RB structure are in compliance with the requirements of ANSI/AISC N690.

#### **9B.2.3.4.6 Pool Liners and Appurtenances**

Construction, fabrication, and installation of metallic liners, non-metallic liners, coatings, joint sealants, and other suitable water tightness measures for structures where leak tightness is critical meet requirements of CSA N291 (Reference 9B.4-34), Clause 6.9.2.

Pool liners are selected to withstand the environmental conditions they are to encounter during the life of the facility. Pool liners, and their connections, are designed to maintain leak tightness and maintain the pool inventory at acceptable levels during and after normal operating, AOO and design basis events.

#### **9B.2.3.5 Interfaces with Other Equipment or Systems**

The RB is in close proximity to the RWB, CB, TB, and Reactor Auxiliary Bay. The interaction of the RB with the adjacent structures under seismic and extreme tornado wind conditions is evaluated per criteria in Chapter 3, Subsections 3.3.1.2 and 3.3.2.8 to ensure these structures will not collapse or collide with the RB and impair the safety functions of Seismic Category A or B SSC.

Adequate gaps are provided between the RB and the surrounding structures to prevent collision between structures. These gaps are evaluated along the entire height of the adjacent structures considering inelastic deformations, construction tolerances, and possible differential settlements as described in Chapter 3, Subsection 3.3.1.2.8.

#### **9B.2.3.6 System and Equipment Operation**

Dynamic mass of the polar crane during the DBE is considered in the seismic analysis. Crane payload analysis will be included in the final design that will be reported in the POSAR in support of the Licence to Operate application.

#### **9B.2.3.7 Instrumentation and Control**

Not applicable to design of RB structure.

#### **9B.2.3.8 Monitoring, Testing, Inspection and Maintenance**

The structural integrity of the RB structure is ensured throughout the life of the plant through monitoring, testing, inspections, and maintenance in compliance with CNSC REGDOC-2.5.2, Section 7.14.

Testing and in-service inspection of the RB structure are performed per the criteria outlined in Chapter 3, Subsection 3.5.5.6.

Field monitoring of the settlement and tilt of the RB shaft is performed as described in Subsection 9B.1.8.

#### **9B.2.3.9 Radiological Aspects**

Design measures that enable the RB to provide flow resistance in contaminated areas that hold up radioactive releases from the containment are discussed in Chapter 12, Subsection 12.3.10. They include the use of a suitable shielding for the fuel pool and associated equipment in addition to using a system for detecting fuel pool leakage.

The design of the SCCV, RPV pedestal and the bioshield, including their selected materials and geometric configuration, ensures that these structures provide a proper shielding from radiation emanating from the fuel core in the RPV.

Shielding of personnel in the SCR is discussed in Chapter 12, Section 12.4.

#### **9B.2.3.10 Performance and Safety Evaluation**

To meet its functional and performance requirements listed in Subsection 9B.2.3.1, the BWRX-300 RB is evaluated and designed for the design loads described in Chapter 3, Subsection 3.5.5.2 and its robustness evaluated for the DEC loads described in Chapter 3, Subsection 3.5.6.

Design basis loads for the RB include loads generated by external and internal hazards identified in Chapter 3, Sections 3.3 and 3.4. Potential combination and interaction of the various hazards on the structure are considered in the design load combinations presented in Subsection 9B.2.3.10.1.

As described in Chapter 9A, Subsection 9A.5.1, the RB HVAC is provided tornado rated dampers at RB penetrations to withstand high wind events and protect its structure in accordance with CNSC REGDOC-2.5.2, Section 7.15.1.

To prevent and resist the spread of internal fires within the building, the RB is supplied with fire detection and suppression systems in compliance with CNSC REGDOC-2.10.1 and CSA N293 and as described in Chapter 3, Subsection 3.4.1.1 They include manual fire alarm boxes installed at each stair door on each level and smoke/fire detectors in various rooms. Smoke and heat detectors are also provided for its service elevator, while occupant notification is provided throughout the building.

Fire protection water for the RB to support automatic fire suppression and manual firefighting is supplied from two separate connections from the site loop. The fire loop feeds the RB on the east side through the exit passageway extending through Plant Service Area (PLSA), and directly into Stair A. The fire loop feeds the RB on the west side through the RWB and CB, to Stair B. A main extends through the RB, between the two stair enclosures, on the south side, for redundancy of supplies. Details associated with fire protection system design for this structure are provided in Chapter 9A, Section 9A.6.

Also, in accordance with Chapter 3, Subsection 3.4.1.1, walls, floors, ceilings, doors, HVAC dampers and penetrations in the RB are fire rated, as required under the fire hazard assessment. Fire barriers are also employed to prevent the spreading of fires from a non-safety room to a room with safety class system or equipment and between redundant safety class fire zones.

For flood protection measures for the RB, refer to Chapter 3, Subsections 3.3.3 and 3.4.2. As mentioned in Chapter 3, Subsection 3.3.3, suitable provisions are provided below flood elevation to ensure the water tightness of the RB and prevent any water ingress from the surrounding soil.

The following sections provide more details on the RB evaluation and design.

##### **9B.2.3.10.1 RB Design Loads and Load Combinations**

In compliance with CNSC REGDOC-2.5.2, Section 7.15.1, the BWRX-300 RB is designed for all applicable load combinations, with due consideration given to the probability of concurrence, loading time history and sequence of loads, as applicable.

Loads and load combinations considered in the design of the RB are described in Chapter 3, Subsection 3.5.5.2 and include loads applied to the SCCV that have an effect on the RB structure due to the common mat foundation, top slab, and other integrating structural components.

Load combinations used for the design of the RB are listed in Table 9B-4.

##### **9B.2.3.10.2 RB Analysis and Design**

Membrane forces, shear forces, and bending moments used in the design of the RB structure are obtained from the set of analysis described in Chapter 3, Subsection 3.5.1.1 performed using the

FE model of the integrated RB structure developed as described in Chapter 3, Subsections 3.3.1.2 and 3.5.1.1. Details of this FE model are presented in Appendix 9B.A. Key seismic analysis results for the integrated RB are provided in Appendix 9B.B, while interaction evaluation results demonstrating how the structural integrity of the surrounding Power Block structures is maintained under design basis seismic and tornado events are provided in Appendix 9B.C. Demands used in the design of key members within the RB are summarized in Appendix 9B.G.

Seismic demands for the design and evaluation of the structure include seismic inertia load demands related to the response of the seismic mass inertia properties assigned in the FE model, the dynamic earth pressure load demands that include SSSI effects with surrounding Power Block structures and foundations and demands from the impulsive portion of the hydrodynamic pressure load (See Appendices 9B.A and 9B.B). Non-seismic demands include demands due static, quasi-static land thermal loads discussed in Chapter 3, Subsection 3.5.1.1 (See Appendix 9B.D).

The circular design of the embedded section of the BWRX-300 RB structure provides significant torsional support for the building. Accidental torsional moment demands at each floor level are computed as described in Chapter 3, Subsection 3.3.1.2.

As mentioned in Chapter 3, Subsection 3.5.2.2, due to its deep embedment, the sliding and overturning stability of the integrated RB is considered satisfied without the need for explicit sliding and overturning stability evaluations.

#### **9B.2.3.10.3 Structural Design Margins**

Safety margins of the RB structure are ensured by satisfying the design basis and beyond design basis requirements in Chapter 3, Subsection 3.5.5.4 to meet the regulatory requirement in CNSC REGDOC-2.5.2, Section 7.15.1. Bounding demand-to-capacity ratios for key structural components within the RB are provided in Appendix 9B.G to demonstrate the adequacy of the design.

The demand-to-capacity ratio plots presented in Appendix 9B.G indicate design margins for the RB structure of at least 15% with load combination representing normal operating conditions being more critical for the design than the abnormal load combination including earthquake loads in combination with containment DBA thermal and pressure loads. The largest demand-to-capacity ratios that are calculated for the below grade portion of the RB exterior wall are mainly associated with earth pressure loads. The earth pressure loads include a conservative estimate of the additional anisotropic pressure load applied on the RB exterior wall that account for the potentially undissipated portion of in-situ pressures locked in the DNNP site rock as a result of past seismic activities in the region.

The plots also indicate that the site-specific seismic demands have a less significant impact on the design of the below grade portion of the RB exterior wall and mat foundation. The seismic loads in combination with the other abnormal condition loads govern the design of the above grade portion of the RB exterior wall and wing walls, with a maximum recorded demand-to-capacity ratio less than 0.8 for these structures. This indicates that the DNNP RB structure has more than 20% safety margins available for the DNNP bounding seismic demands.

#### **9B.2.3.10.4 Robustness**

The robustness of the RB structure is evaluated as discussed in Chapter 3, Subsections 3.3.7.4 and 3.5.6. The RB is designed to resist global failure, perforation, spalling, and fuel intrusion from the regulatory defined threats to protect the containment structure housed within its structure. For more details on the robustness of the RB structure against malevolent acts, refer to the BWRX-300 Security Annex.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Design evaluation results demonstrating the robustness of the RB design per criteria outlined in Chapter 3, Subsection 3.5.6, and against climate change conditions and DEC conditions resulting from a rail derailment accident discussed in Chapter 2, will be reported in the POSAR in support of the Licence to Operate application.



### **9B.3 Other Structures**

#### **9B.3.1 Radwaste Building**

The RWB houses the offgas system charcoal adsorbers, refueling water storage tanks, and rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes.

The RWB location is shown in Chapter 1, Appendix A, Figure A1.4-1. General dimensions of the building are listed in Chapter 1, Table 1.5-2.

The RWB structure consists of reinforced concrete walls and floor slabs supported on a shallow reinforced concrete mat foundation with roof joists and composite roof decking. The lateral force resisting system of the structure consists of concrete shear walls, concrete floors, and a composite roof deck acting as a diaphragm.

Because of its main function which consists of processing and housing liquid, solid and gaseous radioactive wastes, the RWB is categorized as a RW-IIa structure per U.S. NRC RG 1.143 (Reference 9B.4-35). Per Chapter 3, Subsection 3.2.3.1, the RWB is also evaluated for seismic interaction due to its proximity to the RB.

Seismic design bases for the RWB based on its seismic categorization are summarized in Chapter 3, Table 3.3-1. The RWB is designed in accordance with CSA N291 and associated design codes and standards modified by the graded design approach in U.S. NRC RG 1.143. Seismic analysis of the RWB is performed in accordance with CSA N289.3 (Reference 9B.4-36) and ASCE/SEI 4 (Reference 9B.4-37). Seismic interaction evaluation of the RWB follows the requirements in Chapter 3, Subsection 3.3.1.2.8.

The RWB design also considers tornado and missile impact in accordance with requirements in U.S. NRC RG 1.143.

Loads and load combinations used in the design of the RWB are in accordance with CSA N291, with design earthquake and tornado loads applicable to RWB defined per the provisions of U.S. NRC RG 1.143. Environmental load combinations per Table 3 of CSA N291 are considered for interaction evaluations of RWB with RB under DBE and design basis tornado wind conditions.

Design and detailing of the RWB structural steel are in accordance with CSA S16 (Reference 9B.4-38) and in accordance with CSA A23.3 (Reference 9B.4-39) for structural concrete.

The RWB forms part of the combined model for the seismic analysis of the BWRX-300 Power Block structures accounting for SSI and SSSI effects as described in Chapter 3, Subsection 3.3.1.2. The design and seismic interaction evaluation of the RWB structure are performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces. The ASCE/SEI 4 Two-Step analysis approach is used for the seismic qualification of the RWB structure where seismic demands in the form of RWB zero period accelerations or foundation acceleration response spectra are developed from the RB SSI analyses for responses of the RWB simplified dynamic model discussed in Chapter 3, Subsection 3.3.1.2.3. A fixed base quasi-static analysis using the maximum nodal accelerations or response spectrum analysis using the input foundation response spectrum enveloped with the RWB design earthquake is then performed on the standalone refined FE model of the RWB. Resulting forces and stresses are used for the seismic design of the RWB. The demands for the RWB foundation design are taken directly from the RB SSSI analysis, with input corresponding to the RWB design earthquake.

The interaction and structural integrity evaluations are performed for DBE loads using the results of the SSI analysis discussed in Chapter 3, Subsection 3.5.1.1 and for design basis tornado loads. As mentioned in Chapter 3, Subsection 3.3.1.2.8, a clear gap is maintained between the RB and

the RWB to prevent any physical contact during a seismic or tornado event. Clear gaps also separates the RWB from the adjacent CB and TB.

Design features for radiation protection in the RWB are discussed in Chapter 12, Subsection 12.3.9.

As mentioned in Chapter 3, Subsection 3.4.1.1.1, the RWB is designed to have a 3-hour fire resistance rating. Fire protection features for the RWB comply with requirements in CNSC REGDOC-2.10.1 and CSA N293 and include fire alarms and suppression systems as discussed in Chapter 3, Subsection 3.4.1.1. Fire alarm and smoke detectors are installed, with occupant notification provided, throughout the building. Sprinklers and spatial separation are used in various areas of the building to prevent the spread and damage due to fires. Fire protection water supplied to the RWB comes from separate standpipes, one through the RB and one through the CB. Details associated with fire protection system design for this structure are provide in Chapter 9A, Section 9A.6.

#### **9B.3.1.1 Foundation Design Basis**

Geotechnical design of the RWB concrete foundation is in accordance with Canadian Foundation Engineering Manual (Reference 9B.4-40).

Structural design and analysis of the foundation is in accordance with CSA N291, with applicable associated provisions in the National Building Code of Canada (NBC) (Reference 9B.4-41) and CSA A23.3 to ensure the structural integrity of the RWB. Loads, load combinations and acceptance criteria for the RWB foundation design are the same as those used for the design of the super structure.

Stability evaluations of the RWB foundation against sliding and overturning due to DBE, wind, and tornadoes and against floatation due to flooding are performed to ensure the foundation stability requirements in Chapter 3, Subsection 3.5.2.2 are met. The liquefaction potential of the RWB foundation is assessed in Chapter 2, Subsection 2.7.4.7.6 in accordance with Clause 5.6 of CSA N289.3 to ensure a proper siting of the RWB. The RWB foundation rests on overlies material not susceptible to frost action, at a level below which damage due to frosting does not occur in accordance with requirements in ASCE/SEI 32 (Reference 9B.4-42), Canadian Foundation Engineering Manual and NBC Part 4 of Division B (Reference 9B.4-43).

Groundwater effects on the shallow foundation are considered in the settlement analysis and the evaluation of soil capacity as discussed in Chapter 2, Subsection 2.7.3.3.

#### **9B.3.2 Control Building**

The CB houses the Main Control Room (MCR), Emergency Operations Centre, electrical, control, instrumentation equipment, and various other functional areas. The CB location is shown in Chapter 1, Appendix A, Figure A1.4-1. General dimensions of the building are listed in Chapter 1, Table 1.5-2.

As indicated in Chapter 3, Table 3.3-1, the CB is categorized as a Non-Seismic Category structure and is evaluated for seismic interaction due to its proximity to the RB per Chapter 3, Subsection 3.2.3.1. Seismic interaction evaluation of the CB follows the requirements in Chapter 3, Subsection 3.3.1.2.8. The CB is considered a post-disaster building per NBC.

The CB structure consists of a building frame system with perimeter reinforced concrete wall, interior steel columns, beams/girders, roof bar joists, and steel-concrete roof deck as a gravity load carrying system. The CB is founded on reinforced concrete mat foundation. The lateral force resisting system of the structure consists of reinforced concrete shear walls and a steel-concrete roof deck as a diaphragm. A portion of the CB on the east side, consists of a steel frame

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

structure connected to the east perimeter concrete wall. This portion of the structure represents the southern portion of the PLSA as described in Chapter 1, Subsection 1.5.2.3. The lateral force resisting system of this portion consists of reinforced concrete shear wall with parallel braced steel bay on exterior column row and roof deck as a diaphragm.

Due to its proximity to the RB, the CB is designed to maintain its structural integrity under DBE and extreme tornado wind conditions to prevent injuries to control room occupants and to not compromise the structural integrity and safety functions of the adjacent Seismic Category A or B SSC. To ensure the safety of operators in the MCR, the MCR is surrounded by reinforced concrete shear wall. Per Chapter 6, Section 6.4, operators are expected to remain in the MCR and safely operate the plant for most of the Postulated Initiating Events (PIEs). For more details on the conditions that necessitate an evacuation of the MCR, and on travel routes provided to ensure safe egress of operators, refer to Chapter 6, Section 6.4. For special hardening provisions considered in the design of the CB, refer to Subsection 9B.3.2.2.

The CB analysis and design are based on requirements in the NBC for a post-disaster building. Seismic design basis for the CB based on its seismic categorization are summarized in Chapter 3, Table 3.3-1. The CB is also designed as a barrier free structure per NBC Section 3.8.

The CB design meets design requirements stipulated in the NBC Part 4 of Division B. Loads and load combinations used in the design of the CB are in accordance with the NBC. Environmental load combinations per Table 3 of CSA N291 are considered for interaction evaluations of CB with RB under DBE and design basis tornado wind conditions.

Design and detailing of the CB structural steel is in accordance with CSA S16 and in accordance with CSA A23.3 for structural concrete.

The CB forms part of the combined model for the seismic analysis of the BWRX-300 Power Block structures accounting for SSI and SSSI effects as described in Chapter 3, Subsection 3.3.1.2. For the NBC design and seismic interaction evaluation, analysis and design of the CB structure is performed following linear elastic analysis methodology.

The seismic analysis and design of the CB structure is performed in accordance with the NBC using a site-specific Non-Seismic Design Basis Earthquake (NS-DBE). This site-specific NS-DBE is defined based on 2% probability of exceedance in 50 years (i.e., the uniform hazard response spectra with return period of 2475 years or annual probability of exceedance of  $4E-4$ ) as specified in Section 4.1.8.4 of the NBC. Ground motion parameters are determined based on the site designation at the DNNP site. Ground motion time histories used as input in the dynamic time history analysis are selected and scaled to be compatible with the NS-DBE per the guidelines in the NBC, Part 4 of Division B. Additionally, the CB is verified using a 5%-damped input motion based on a 5% probability of exceedance in 50 years in accordance with NBC, Section 4.1.8.23 Sentence (2), as applicable. Non-Seismic structures are not explicitly analysed and designed to resist vertical ground motions consistent with the NBC.

The interaction and structural integrity evaluations are performed for DBE using the results of the SSI analysis discussed in Chapter 3, Subsection 3.5.1.1, and design basis tornado loads. As mentioned in Chapter 3, Subsection 3.3.1.2.8, a clear gap is maintained between the RB and the CB to prevent any physical contact during a seismic or tornado event. Clear gaps also separates the CB from the adjacent RWB and Reactor Auxiliary Bay.

Design features for radiation protection in the CB are discussed in Chapter 12, Subsection 12.3.11. Design features for habitability of the MCR are described in Chapter 6, Section 6.4.

As mentioned in Chapter 3, Subsection 3.4.1.1.1, the CB is designed to have a 3-hour fire resistance rating. Fire protection features in the CB comply with requirements in CNSC REGDOC-

2.10.1 and CSA N293 and include fire alarms and suppression systems as described in Chapter 3, Subsection 3.4.1.1. Fire alarm and smoke detectors are installed, with occupant notification provided, throughout the building. Sprinklers are installed within the building, with exception of the Control Room where portable fire extinguishers are provided. Fire protection water supplied to the building comes from two directions, one through the RWB and one from the site loop on the south side of the building. Details associated with fire protection system design for this structure are provide in Chapter 9A, Section 9A.6.

#### **9B.3.2.1 Foundation Design Bases**

Design bases for the CB foundation are similar to those for the RWB foundation discussed in Subsection 9B.3.1.1.

Stability evaluations of the CB foundation against sliding and overturning due to DBE, wind, and tornadoes and against floatation due to flooding are performed to ensure the foundation stability requirements in Chapter 3, Subsection 3.5.2.2 are met. Potential strata liquefaction under the CB foundation is assessed in Chapter 2, Subsection 2.7.4.7.6 in accordance with Clause 5.6 of CSA N289.3 to ensure a proper siting of the CB. The CB foundation rests on overlies material not susceptible to frost action, at a level below which damage due to frosting does not occur in accordance with requirements in ASCE/SEI 32, Canadian Foundation Engineering Manual and NBC Part 4 of Division B.

Groundwater effects on the shallow foundation are considered in the settlement analysis and the evaluation of soil capacity as discussed in Chapter 2, Subsection 2.7.3.3.

#### **9B.3.2.2 Special Hardening Provision for Control Room Habitability and Post-Accident Response Requirements**

In accordance with CNSC REGDOC-2.5.2, Section 8.10.1, to ensure the habitability of the MCR and post-accident requirements, the CB is hardened by design for the full tornado (1E-7 per year return period) effect. Area(s) required to be hardened are to be designated during design development.

Structures of hardened areas are designed to remain functional before, during and after the design tornado event. Interaction criteria discussed in Chapter 3, Subsections 3.2.3.1 and 3.3.1.2.8 apply under this loading event.

As discussed in Chapter 9A, Subsection 9A.5.2, the CB HVAC is provided tornado rated dampers as necessary to withstand high wind events in accordance with CNSC REGDOC-2.5.2, Section 7.15.1.

#### **9B.3.3 Turbine Building**

The TB encloses the turbine generator, main condenser, condensate and feedwater systems, condensate purification system, offgas catalytic recombiner, cooling condenser, refrigerant dryer, turbine generator support system and bridge crane. Per Table 3.3-1 of Chapter 3, the TB is categorized as a Non-Seismic Category structure and is evaluated for seismic interaction, similar to the CB. The TB is considered a post-disaster building per NBC.

The layout and key dimensions of the TB are shown in Chapter 1, Appendix A, Figure A1.4-1 and Chapter 1, Table 1.5-2, respectively.

The TB structure is divided into three structural supporting systems:

1. The TB shell structure which consists of a steel frame system with steel columns, beams/girders, roof bar joists, and floor/roof decks as gravity load carrying systems. The lateral force resisting system consists of braced frames and floor/roof decks as

diaphragms. The low-rise steel frame structure on the east side of the TB connecting to, and integrated with, its eastern shell structure side represents the northern portion of the PLSA as discussed in Chapter 1, Subsection 1.5.2.2. The lateral force resisting system of this portion consists of parallel braced frame on exterior column row and roof deck as a diaphragm.

2. The shielding wall area which consists of reinforced concrete shear walls surrounding and supporting the portion of the structure containing radioactive steam and water for shielding (turbines, condenser, offgas vessels, heater tanks, and associated piping). This area includes some gravity load carrying internal steel columns and is isolated from the TB steel structure, despite sharing a common mat foundation with the TB shell structure.
3. The reinforced concrete pedestal supporting the turbine, generator, and exciter within the TB shell structure and the shielding walls area. This turbine generator foundation is structurally isolated from both the shielding walls and the TB shell structure for vibration control. However, its dominant dynamic characteristics are included in the SSSI model.

The analysis and design approaches and requirements for the TB and its foundation, including the interaction evaluations for DBE or design basis tornado loads, are the same as those for the CB discussed in Subsection 9B.3.2, excluding the special hardening provisions. The turbine generator foundation is sized and designed to additionally meet the applicable static and dynamic performance requirements by the turbine generator manufacturer, and to have adequate structural integrity and stability to withstand applicable loads and associated load combinations imposed during the service life of the plant. Local hardening is provided for systems in the TB as required. Like the CB, the TB is also separated from the RB by a clear gap to prevent any physical contact during a seismic or tornado event. Clear gaps also separates the TB from the adjacent RWB and Reactor Auxiliary Bay.

The TB is also designed to resist damage and the spread of fires as discussed in Chapter 3, Subsection 3.4.1.1 and in compliance with CNSC REGDOC-2.10.1 and CSA N293. Fire alarm and smoke detectors are installed, with occupant notification provided, throughout the building. Areas housing the turbine generator and associated ancillary process equipment and electrical equipment rooms in the building are designed to have a 3-hour fire resistance rating. Fireproofing of structural steel members under design basis turbine lube oil fire is as discussed in Chapter 3, Subsection 3.4.1.1.

To satisfy CSA N293 requirements, sprinklers are provided in the turbine generator underfloor area below the operating floor, in the steam turbine shielded area, in the freight elevator hoistway and in the machine room. Lube oil system continuing out of the shielded area go to the bearings associated with the generator.

The largest flow rate is anticipated to be the simultaneous operation of a turbine underfloor system (both levels) and the turbine bearing system, including hose streams. Methodology used to calculate this flow is per CSA N293 and CSA N293S1.

Fire protection water is supplied to the TB from three separate connections from the site loop which feed into Stairs C, B and A.

Radiological aspects of the TB are discussed in Chapter 12, Subsections 12.2.1.3 and 12.2.2.2.

#### **9B.3.4 Reactor Auxiliary Bay**

The Reactor Auxiliary Bay is the independent structure east of the RB shown in Chapter 1, Appendix A, Figure A1.4-1. As indicated in Chapter 3, Table 3.3-1, the Reactor Auxiliary Bay is categorized as a Non-Seismic Category structure and is evaluated for seismic interaction, similar to the CB and TB. The Reactor Auxiliary Bay is considered a post-disaster building per NBC.

The Reactor Auxiliary Bay structure consists of a one-story structural steel building with a combination of moment frame and braced frame systems in the transverse (short) direction and braced bays in the longitudinal direction, with the roof deck acting as the diaphragm. The Reactor Auxiliary Bay rests on a shallow foundation isolated from the adjacent building foundations. The roof elevation of the Reactor Auxiliary Bay is flush with the CB roof.

The analysis and design approaches and requirements for the Reactor Auxiliary Bay and its foundation, including the interaction evaluations for DBE and design basis tornado loads, are the same as those for the CB and TB discussed in Subsections 9B.3.2 and 9B.3.3. A clear gap separates the Reactor Auxiliary Bay from the adjacent RB to prevent any physical contact between the buildings during a seismic or tornado event.

Radiological aspects of the Reactor Auxiliary Bay are discussed in Chapter 12, Subsection 12.2.2.5.

Fire protection design features in this area are as described in Chapter 3, Subsection 3.4.1.1 and include the use of wet pipe sprinklers. As mentioned in Chapter 3, Subsection 3.4.1.1.1, the Reactor Auxiliary Bay is designed to have a 3-hour fire resistance rating.

### **9B.3.5 Pumphouse/Forebay and Tunnels**

#### **9B.3.5.1 Pumphouse/Forebay**

The Pumphouse/Forebay structure is composed of the Forebay, pump bays, and superstructures housing the Circulating Water System (CWS) pumps and related electrical equipment. For the location of the Pumphouse/Forebay structure, refer to Chapter 1, Appendix A, Figure A1.4-1.

The Forebay provides the transition for the water flowing from the intake tunnel to the pump bays. Pump bays are provided for the large CWS pumps supplying water to the main condenser and the smaller CWS pumps that provide water to the Plant Cooling Water System (PCW) heat exchangers. The top elevation of the pump bay foundation is based on pump minimum submergence depth, minimum liquid depth, and low water level of the lake.

To prevent the intake of biofouling material and mitigate the probability of clogging, design considerations regarding the intake tunnel and lakebed intake structure depths in Lake Ontario are employed. At the same time, traveling screens in the Pumphouse/Forebay are used to mitigate clogging probability. If the lakebed intake structure screens are deemed insufficient for biofouling material intake prevention during the design phase, a fish return system, or slew, may be provided to safely return large fish to Lake Ontario to meet DNNP regulatory commitment D-C-1 (Reference 9B.4-44).

#### **9B.3.5.2 Tunneling**

##### **9B.3.5.2.1 Vertical Shafts**

The onshore vertical shafts are designed to facilitate the operation of up to four Small Modular Reactor (SMR) units and facilitate the construction of the intake and discharge tunnels. The intake tunnel connects to the onshore vertical intake shaft to convey cooling water to the Pumphouse/Forebay. Similarly, the onshore vertical discharge shaft connects to the discharge tunnel to convey heated water through the discharge tunnel to the diffusers.

The depth of the onshore vertical shafts is based on the required depth of the intake and discharge tunnels. The required depth of the tunnels is based on anticipated subsurface conditions below the lake determined by the offshore geotechnical investigation discussed in Chapter 2, Subsection 2.5.2.1.

#### **9B.3.5.2.2 Intake Tunnel**

The intake tunnel conveys cooling water from the lakebed intake structure to the onshore intake vertical shaft. The intake tunnel is located beneath Lake Ontario and is to be lined with concrete.

#### **9B.3.5.2.3 Discharge Tunnel and Diffusers**

The discharge tunnel conveys the discharge water from the onshore discharge vertical shaft to the discharge risers/diffusers. The discharge tunnel is located beneath Lake Ontario and the tunnel is to be lined with concrete.

The diffuser arrangement is to be finalized after completion of thermal plume modeling during design.

#### **9B.3.5.3 Design Considerations**

There are no Safety Category 1 functions provided by the BWRX-300 water intake structures and associated systems and components. The PCW and CWS housed in the Pumphouse/Forebay are SC3 systems as indicated in Chapter 3, Appendix 3A, Table 3.12-1. As a result, the Pumphouse/Forebay is categorized as a Non-Seismic structure and is designed in accordance with the NBC.

#### **9B.3.6 Fire Pump Enclosure**

As shown in Chapter 1, Appendix A, Figure A1.4-1, the fire pumps are housed in an enclosure located within the protected area of the plant. A description of the Fire Pump Enclosure structure and design is covered in the POSAR in support of the Licence to Operate application.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**9B.4 References**

- 9B.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9B.4-2 NEDO-33914 "BWRX-300 Advanced Civil Construction and Design Approach," . GE-Hitachi Nuclear Energy Americas, LLC.
- 9B.4-3 CSA N289.5, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities," CSA Group.
- 9B.4-4 NEDC-33926P, "Steel-Plate Composite Containment Vessel and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.
- 9B.4-5 ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," American Institute of Steel Construction.
- 9B.4-6 ASME "Boiler and Pressure Vessel Code," Section III, Division 1, Subsection NE, Class MC Components," American Society of Mechanical Engineers.
- 9B.4-7 USNRC Regulatory Guide 1.54, "Service Level I, II, III, and In-Scope License Renewal Protection Coatings Applied to Nuclear Power Plants."
- 9B.4-8 ASTM D5144, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants," American Society for Testing and Materials.
- 9B.4-9 ASTM A738/A738M, "Standard Specification for Pressure Vessel Plates, Heat-Treated, Carbon-Manganese-Silicon Steel, for Moderate and Lower Temperature Service," American Society for Testing and Materials.
- 9B.4-10 Not Used
- 9B.4-11 Not Used
- 9B.4-12 ASTM A108, "Standard Specification for Steel Bar, Carbon and Alloy, Cold-Finished," American Society for Testing and Materials.
- 9B.4-13 ASME "Boiler and Pressure Vessel Code (BPVC), Section III: Division 2, Subsection CC, Code for Concrete Containments," American Society of Mechanical Engineers.
- 9B.4-14 ASTM A572/A572M, "Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel," American Society for Testing and Materials.
- 9B.4-15 ASME Code Case N-632, "Use of ASTM A 572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners," American Society of Mechanical Engineers.
- 9B.4-16 ASTM A516/A516M, "Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service," American Society for Testing and Materials.
- 9B.4-17 ASTM A537/A537M, "Standard Specification for Pressure Vessel Plates, Heat Treated, Carbon-Manganese-Silicon Steel," American Society for Testing and Materials.
- 9B.4-18 ASTM A240/A240M, "Standard Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels and for General Application," American Society for Testing and Materials.
- 9B.4-19 ASTM A264, "Standard Specification for Stainless Chromium-Nickel Steel-Clad Plate," American Society for Testing and Materials



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

- 9B.4-20 ASTM A193, "Standard Specification for Alloy-Steel and Stainless Steel Bolting for High-Temperature or High Pressure Service and Other Special Purpose Applications," American Society for Testing and Materials.
- 9B.4-21 ASTM A437, "Standard Specification for Stainless and Alloy Steel Turbine-Type Bolting Specially Heat Treated for High-Temperature Service," American Society for Testing and Materials.
- 9B.4-22 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 9B.4-23 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 9B.4-24 CSA N293, "Fire Protection for Nuclear Power Plants," CSA Group.
- 9B.4-25 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.
- 9B.4-26 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,"
- 9B.4-27 USNRC Regulatory Guide 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, U.S. Nuclear Regulatory Commission."
- 9B.4-28 USNRC Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants."
- 9B.4-29 ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," American Concrete Institute.
- 9B.4-30 ASTM F3125, "Standard Specification for High Strength Structural Bolts and Assemblies, Steel and Alloy Steel, Heat Treated, 120 ksi (830MPa) and 150 ksi (1040 MPa) Minimum Tensile Strength, Inch and Metric Dimensions," American Society for Testing and Materials.
- 9B.4-31 ASTM A354, "Standard Specification for Quenched and Tempered Alloy Steel Bolts, Studs, and Other Externally Threaded Fasteners," American Society for Testing and Materials.
- 9B.4-32 ASTM A449, "Standard Specification for Hex Cap Screws, Bolts and Studs, Steel, Heat Treated, 120/105/90 ksi Minimum Tensile Strength, General Use," American Society for Testing and Materials.
- 9B.4-33 ASTM A307, "Standard Specification for Carbon Steel Bolts, Studs, and Threaded Rod 60 000 PSI Tensile Strength," American Society for Testing and Materials.
- 9B.4-34 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 9B.4-35 USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- 9B.4-36 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 9B.4-37 ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

- 9B.4-38 CSA S16, "Design of Steel Structures," CSA Group.
- 9B.4-39 CSA A23.3, "Design of Concrete Structures," CSA Group.
- 9B.4-40 Canadian Foundation Engineering Manual, 4th Edition," Canadian Geotechnical Society.
- 9B.4-41 Canadian Commission on Building and Fire Codes, National Resource Council of Canada, National Building Code of Canada (NBC).
- 9B.4-42 ASCE/SEI 32, "Design and Construction of Frost-Protected Shallow Foundations," American Society of Civil Engineers.
- 9B.4-43 Canadian Commission on Building and Fire Codes, National Resource Council of Canada, National Building Code of Canada (NBC), Part 4 of Division B, User's Guide Structural Commentaries, 4th Edition.
- 9B.4-44 NK054-REP-01210-00078 R007, Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 9B.4-45 USNRC Regulatory Guide 1.57, "Design Limits and Load Combinations for Metal Primary Reactor Containment System Components."
- 9B.4-46 USNRC Regulatory Guide 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments."
- 9B.4-47 S. Shafaei, A. H. Varma, J. Seo, D. Huber, and R. Klemencic, "Wind Design of Composite Plate Shear Walls/Concrete Filled (SpeedCore) Systems," Engineering Journal, 2022, 183-208.
- 9B.4-48 EN 1993-1-8: 2005, "Eurocode 3, Design of Steel Structures, Part 1-8, Design of Joints", European Committee for Standardization.
- 9B.4-49 Sarraj, M., "The Behavior of Steel Fin Plate Connections in Fire", Ph.D. Thesis, Department of Civil and Structural Engineering, University of Sheffield, UK, 2007.
- 9B.4-50 USNRC DC/COL-ISG-017, "Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses."
- 9B.4-51 USNRC Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants."
- 9B.4-52 ANSI/AISC 360-16, Specification for Structural Steel Buildings, American Institute of Steel Construction.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B-1: Load Combinations and Load Factors for SCCV**

Load Category	Combination #	Loads																		Load Condition
		D	F	L	H	P <sub>t</sub>	T <sub>t</sub>	P <sub>v</sub> /P <sub>o</sub>	R <sub>o</sub>	T <sub>o</sub>	W	E <sub>s</sub>	W <sub>t</sub>	P <sub>a</sub>	R <sub>a</sub>	T <sub>a</sub>	Y <sub>j</sub> /R <sub>rj</sub>	Y <sub>m</sub> /R <sub>rm</sub>	Y <sub>r</sub> /R <sub>rr</sub>	
Test	SCCV-1	1.0	1.0	1.0	1.0	1.0	1.0													Service
Construction	SCCV-2	1.0	1.0	1.0	1.0					1.0	1.0									
Normal	SCCV-3	1.0	1.0	1.0	1.0			1.0	1.0	1.0										
Severe Environmental	SCCV-4	1.0	1.0	1.30	1.30			1.0	1.0	1.0	1.5									Factored
Extreme Environmental	SCCV-5	1.0	1.0	1.0	1.0			1.0	1.0	1.0		1.0								
	SCCV-6	1.0	1.0	1.0	1.0			1.0	1.0	1.0			1.0							
Abnormal	SCCV-7	1.0	1.0	1.0	1.0									1.5	1.0	1.0				
	SCCV-8	1.0	1.0	1.0	1.0									1.0	1.25	1.0				
	SCCV-9	1.0	1.0	1.0	1.0									1.25	1.0	1.0				
Abnormal / Extreme Environmental	SCCV-10	1.0	1.0	1.0	1.0						1.25			1.25	1.0	1.0				
	SCCV-11	1.0	1.0	1.0	1.0						-	1.0		1.0	1.0	1.0	1.0	1.0	1.0	

- (1) The load combination involving post-LOCA flooding condition with OBE (set at 1/3 of the DBE, see Chapter 3, Subsection 3.3.1) is eliminated in accordance with U.S. NUREG-0800 SRP 3.8.1. This load combination is bounded by SCCV-11 since the maximum hydrostatic pressure produced by the post-LOCA flooding is less significant than the accident LOCA pressure.

D = Dead Loads

F = Hydrostatic Pressure Loads

L = Live Loads

H = Soil and Underground Water Pressure Loads

P<sub>t</sub> = Test Pressure Loads

T<sub>t</sub> = Test Thermal Loads

P<sub>v</sub>/P<sub>o</sub> = Pressure Variant Load

R<sub>o</sub> = Normal Operating Pipe Reaction Loads

T<sub>o</sub> = Normal Operating Thermal Loads

W = Wind Loads

E<sub>s</sub> = Seismic Loads (generated by DBE)

W<sub>t</sub> = Tornado Loads

P<sub>a</sub> = Accident Pressure Loads

R<sub>a</sub> = Accident Pipe Reaction Loads

T<sub>a</sub> = Accident Thermal Loads

Y<sub>j</sub> / R<sub>rj</sub> = Jet Impingement Loads

Y<sub>m</sub> / R<sub>rm</sub> = Missile Impact Loads

Y<sub>r</sub> / R<sub>rr</sub> = High Energy Pipe Rupture

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B-2: Load Combinations and Load Factors for Containment Closure Head**

Service Level	No.	Load Combination									
		D	L	E <sub>s</sub>	T <sub>o</sub>	T <sub>a</sub>	T <sub>t</sub>	P <sub>o</sub>	P <sub>a</sub>	P <sub>t</sub>	R <sub>r</sub>
Test Condition	1	1	1				1			1	
Design Condition	2	1	1			1			1		
Level A	3	1	1		1			1			
	4	1	1			1			1		
	5	1	1		1			1			
	6	1									
Level C	7	1	1	1		1			1		
	8	1	1	1	1			1			
	9	1									
	10	1									
Level D	11	1	1	1		1			1		1

(1) Based on RG 1.57 (Reference 9B.4-45) and U.S. NUREG-0800, SRP 3.8.2

(2) D = Dead Loads

L = Live Loads

E<sub>s</sub> = Seismic Loads (generated by DBE)

T<sub>o</sub> = Normal Operating Thermal Loads

T<sub>a</sub> = Accident Thermal Loads

T<sub>t</sub> = Test Thermal Loads

P<sub>o</sub> = Normal Operating Pressure Loads including Pressure Variant Loads (Pv)

P<sub>a</sub> = Accident Pressure Loads

P<sub>t</sub> = Test Pressure Loads

R<sub>r</sub> = Local Effects on Containment due to LOCA (R<sub>rj</sub> + R<sub>rm</sub>)

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B-3: Load Combinations and Load Factors for Other Class MC Components**

Service Level	No.	Load Combination												
		D	L	E <sub>s</sub>	T <sub>o</sub>	T <sub>a</sub>	T <sub>t</sub>	P <sub>o</sub>	P <sub>a</sub>	P <sub>t</sub>	R <sub>r</sub>	R <sub>o</sub>	R <sub>a</sub>	H <sub>a</sub>
Test Condition	1	1	1				1			1				
Design Condition	2	1	1			1			1				1	
Level A	3	1	1		1			1				1		
	4	1	1			1			1				1	
	5	1	1		1			1				1		
	6	1												
Level C	7	1	1	1		1			1				1	
	8	1	1	1	1			1				1		
	9	1												
	10	1												
Level D	11	1	1	1		1			1		1		1	
Post-flooding Condition	12	1	1	1 <sup>2</sup>										1

(1) Based on RG 1.57 and U.S. NUREG-0800, SRP 3.8.2

(2) OBE consideration only for post-flooding condition and cyclic loading considerations, in accordance with U.S. NUREG-0800, SRP 3.8.2

(3) D = Dead Loads

L = Live Loads

E<sub>s</sub> = Seismic Loads (generated by DBE)

T<sub>o</sub> = Normal Operating Thermal Loads

T<sub>a</sub> = Accident Thermal Loads

T<sub>t</sub> = Test Thermal Loads

P<sub>o</sub> = Normal Operating Pressure Loads including Pressure Variant Loads (Pv)

P<sub>a</sub> = Accident Pressure Loads

P<sub>t</sub> = Test Pressure Loads

R<sub>r</sub> = Local Effects on Containment due to LOCA (R<sub>rr</sub> + R<sub>rj</sub> + R<sub>rm</sub>)

R<sub>o</sub> = Normal Operating Pipe Reaction Loads

R<sub>a</sub> = Accident Pipe Reaction Loads

H<sub>a</sub> = Internal Flooding Loads

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B-4: Load Combinations, Load Factors and Acceptance Criteria for RB Safety  
Class Steel Bricks™ Modules and Steel Structures**

Load Category	Combination #	Loads																				
		D	F	L	H	L <sub>r</sub>	R	S	C	R <sub>o</sub>	P <sub>o</sub>	T <sub>o</sub>	W	E <sub>s</sub>	W <sub>t</sub>	S <sub>x</sub>	P <sub>a</sub>	R <sub>a</sub>	T <sub>a</sub>	Y <sub>j</sub>	Y <sub>m</sub>	Y <sub>r</sub>
Normal	RB-1	1.4	1.4						1.0	1.4	1.0	1.0										
	RB-2.1	1.2	1.2	1.6	1.6	0.5			1.4	1.2	1.2	1.2										
	RB-2.2	1.2	1.2	1.6	1.6		0.5		1.4	1.2	1.2	1.2										
	RB-2.3	1.2	1.2	1.6	1.6			0.5	1.4	1.2	1.2	1.2										
	RB-3.1	1.2	1.2	0.8	0.8	1.6			1.4	1.2	1.2	1.2										
	RB-3.2	1.2	1.2	0.8	0.8		1.6		1.4	1.2	1.2	1.2										
	RB-3.3	1.2	1.2	0.8	0.8			1.6	1.4	1.2	1.2	1.2										
	RB-2/3	1.2	1.2	1.6	1.6			1.6	1.4	1.2	1.2	1.2										
Severe Environmental	RB-4.1	1.2	1.2	1.6	1.6	0.5			1.0	1.2	1.0	1.0	1.0									
	RB-4.2	1.2	1.2	1.6	1.6		0.5		1.0	1.2	1.0	1.0	1.0									
	RB-4.3	1.2	1.2	1.6	1.6			0.5	1.0	1.2	1.0	1.0	1.0									
Extreme Environmental	RB-5	1.0	1.0	0.8	1.0				1.0	1.0	1.0	1.0		1.0								
	RB-6	1.0	1.0	0.8	1.0					1.0	1.0	1.0			1.0							
	RB-7	1.0	1.0	0.8	1.0					1.0	1.0	1.0				1.0						
Abnormal	RB-8	1.0	1.0	0.8	1.0				1.0								1.4	1.0	1.0			
	RB-9	1.0	1.0	0.8	1.0									0.7			1.0	1.0	1.0	1.0	1.0	1.0

- (1) Load combinations are in accordance with ANSI/AISC N690 and U.S. NRC RG 1.243 (Reference 9B.4-46). Load combination RB2/3 is a conservative bounding load combination for RB-2.1, RB-2.2, RB-2.3, RB-3.1, RB-3.2, and RB-3.3 required by ANSI/AISC N690.

D = Dead Loads

F = Hydrostatic Pressure Loads

L = Live Loads

H = Soil and Underground Water Pressure Loads

L<sub>r</sub> = Roof Live Loads

R = Rain Loads

S = Snow Loads

C = Crane Loads

R<sub>o</sub> = Normal Operating Pipe Reactions

P<sub>o</sub> = Normal Operating Pressure Loads

T<sub>o</sub> = Normal Operating Thermal Loads

W = Wind Loads

E<sub>s</sub> = Seismic Loads (Generated by DBE)

W<sub>t</sub> = Tornado Wind Loads

S<sub>x</sub> = Extreme Snow Loads

P<sub>a</sub> = Accidental Pressure Loads

R<sub>a</sub> = Accidental Piping Reactions

T<sub>a</sub> = Accidental Thermal Loads

Y<sub>j</sub> = Jet Impingement Loads

Y<sub>m</sub> = Missile Impact Loads

Y<sub>r</sub> = High Energy Pipe Rupture

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

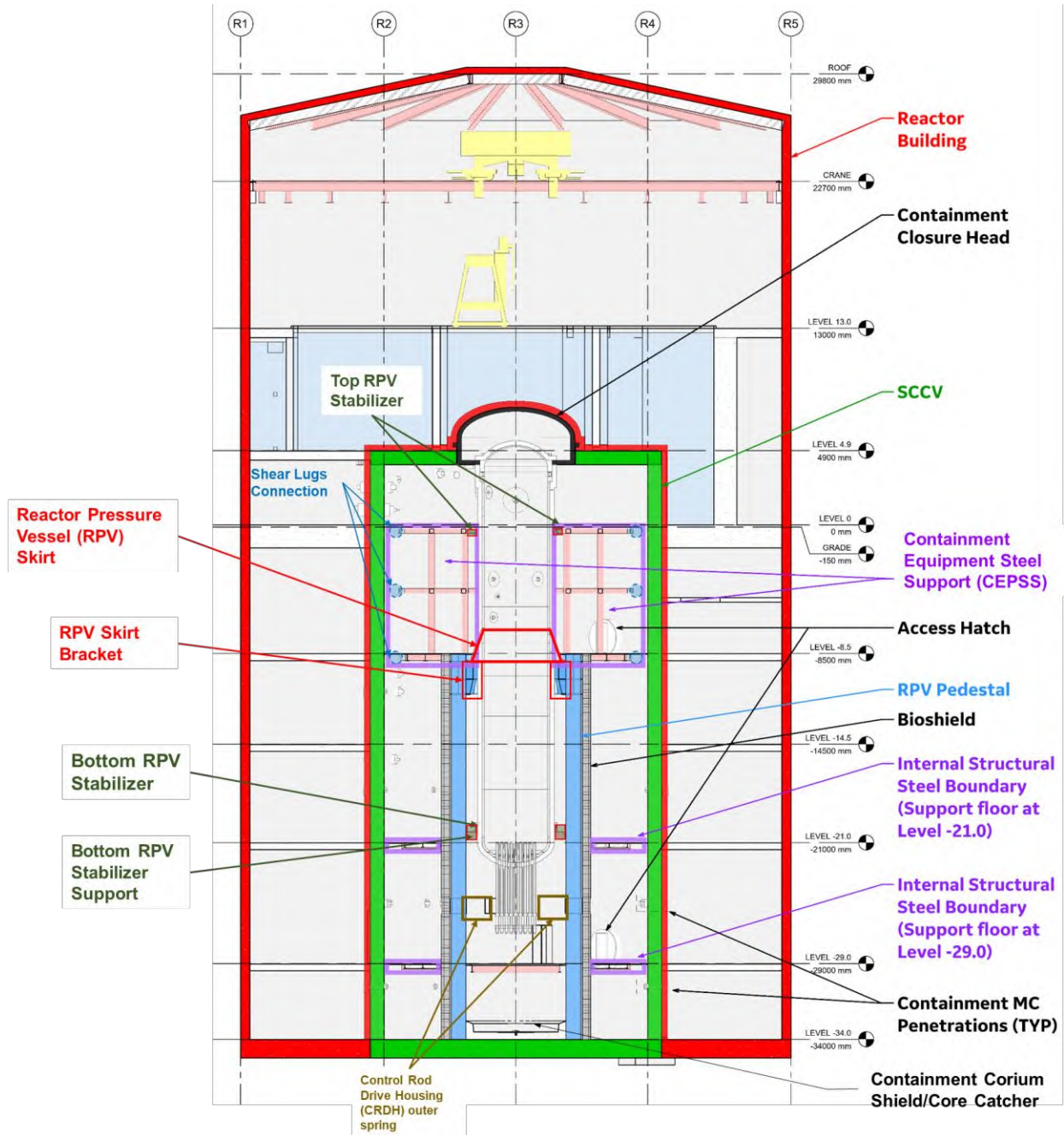


Figure 9B-1: BWRX-300 Integrated RB Structure Including Common Mat Foundation

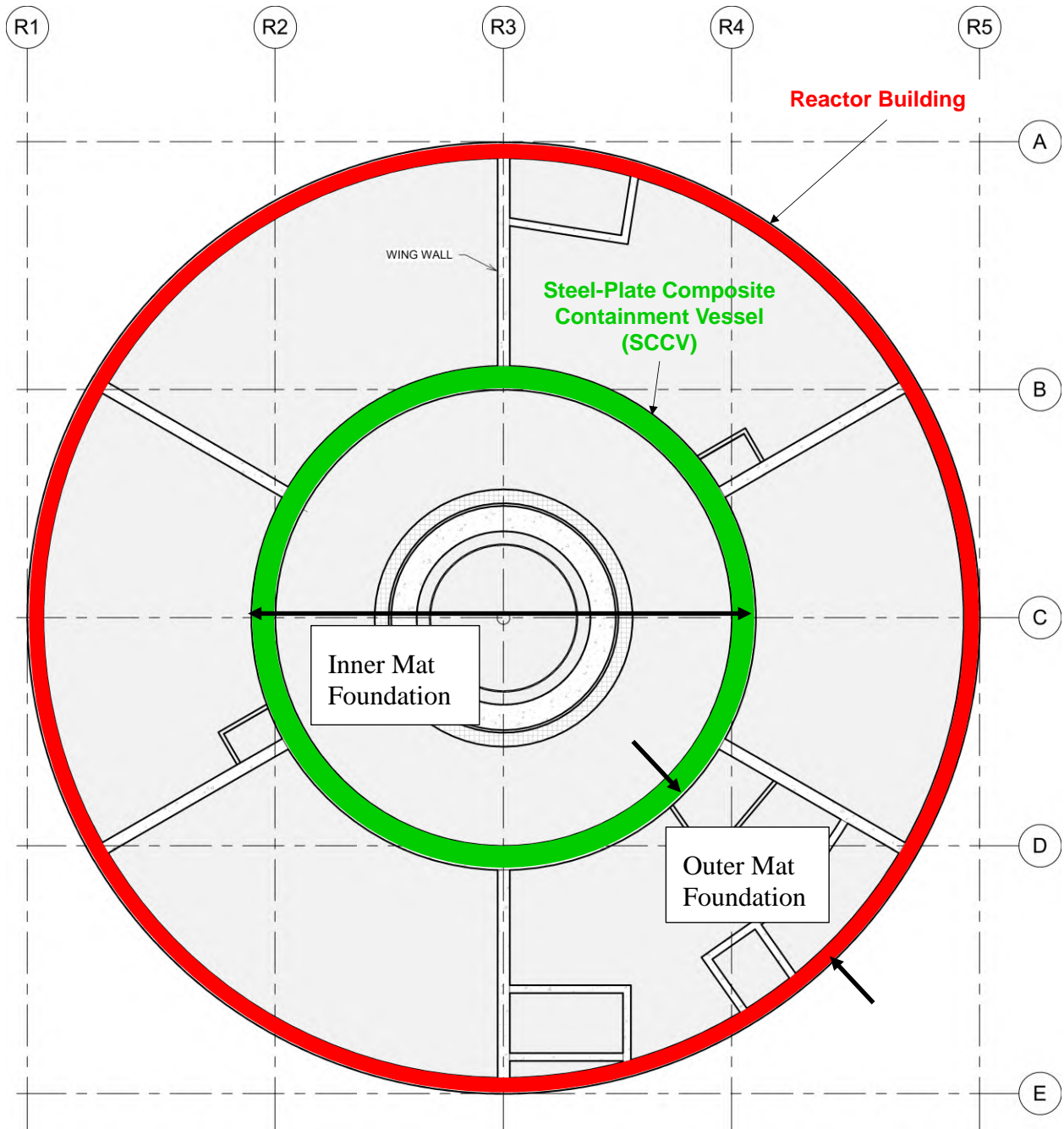
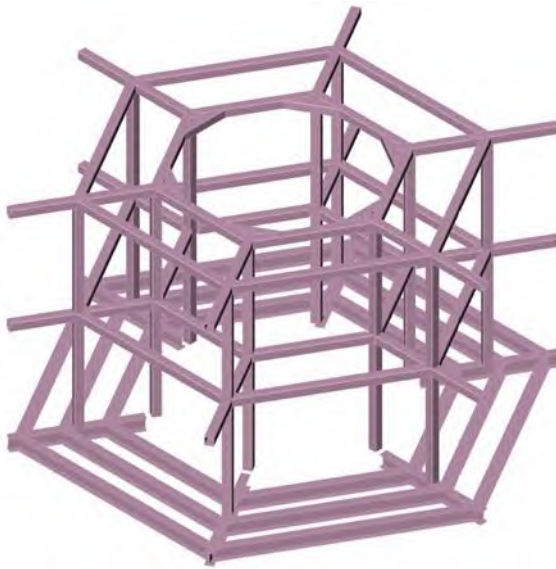
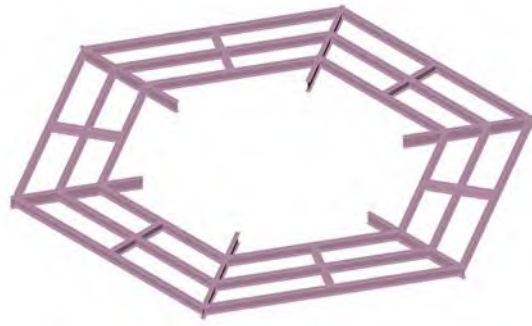


Figure 9B-2: Plan View of BWRX-300 Common Mat Foundation





(a) CEPSS



(b) Support floor at Level -21 and -29 m

**Figure 9B-3: BWRX-300 Containment Internal Structural Steel**

**9B.5 Appendices**

- APPENDIX 9B.A – FINITE ELEMENT MODEL USED IN STRUCTURAL ANALYSES OF INTEGRATED REACTOR
- APPENDIX 9B.B – SEISMIC SOIL-STRUCTURE INTERACTION ANALYSIS
- APPENDIX 9B.C – INTERACTION EVALUATIONS RESULTS
- APPENDIX 9B.D – RESPONSES TO STATIC LOADS
- APPENDIX 9B.E – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT
- APPENDIX 9B.F – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT INTERNAL STRUCTURES
- APPENDIX 9B.G – DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE

## **APPENDIX 9B.A – FINITE ELEMENT MODEL USED IN STRUCTURAL ANALYSES OF INTEGRATED REACTOR**

### **9B.6 Introduction**

This appendix presents the 3-Dimensional (3-D) FE models developed for the analysis and design of the integrated RB structures, which comprise the RB, containment, containment internal structures and their common mat foundation. Methodology used to develop these models are presented in Chapter 3, Subsections 3.3.1.2.2 and 3.5.1.1.1.

The integrated RB FE model used to perform the various structural analyses discussed in Chapter 3, Subsection 3.5.1.1 is presented in Subsection 9B.6.1. This model is developed using appropriate element types to represent the mass, stiffness, damping characteristics of the integrated RB structures for the analysed loads being evaluated and resulting stress responses. The model has a sufficiently refined mesh to accurately capture the dynamic response of the structures over the frequency range of interest and to enable accurate calculation of structural stress demands. Mesh refinements in areas of geometric structural discontinuities are used, as applicable.

Table 9B.6-1 summarizes the integrated RB modeling requirements for the various analysis cases performed. Subsection 9B.6.2 presents the 3-D FE model used for the seismic SSI analysis of the integrated RB. Subsection 9B.6.3 presents the 3-D FE model used for the 1-g static SSI analyses. Subsection 9B.6.4 presents the 3-D FE models used for static and quasi-static analyses, while Subsection 9B.6.5 presents the 3-D FE models used for thermal stress analysis.

### **9B.6.1 Integrated Reactor Building Finite Element Model**

#### **9B.6.1.1 Model Overview**

The integrated RB FE model is developed using ANSYS (See Chapter 3, Appendix 3B for description) and represents the major structural elements of the integrated RB structures as described in Chapter 3, Subsection 3.3.1.2. The origin of the model is defined at the centre of the integrated RB structure with the Z-origin defined at grade (Level 0 m) and the global X- and Y-axes pointed in the South, and East directions, respectively.

The integrated RB FE model is developed using wall centrelines and approximate centrelines of slabs. Slab centrelines used in the integrated RB model are approximations of the weighted average centreline of slabs with varying thicknesses.

Figure 9B.6-1 is a 2/3 slice of the integrated RB FE model showing the major structural components of the model. Containment mechanical and electrical piping penetrations are incorporated in the integrated RB FE model as shown in Figure 9B.6-2.

The RPV and internals, including fuel, valve mass and supports, are integrated into the integrated RB FE model using a Lumped Mass Stick (LMS) model developed as described in Chapter 3, Subsection 3.3.1.3.3 and shown in Figure 9B.6-1. Subsection 9B.6.1.4 discusses the connections used to integrate the LMS and the integrated RB models.

#### **9B.6.1.2 Model Properties**

Material properties used in the construction of the integrated RB are described in Subsection 9B.2. Properties assigned to the integrated RB FE model reflect these properties and are computed in accordance with the guidelines of ANSI/AISC N690.

Response Level 1 and Level 2 damping values applicable to the integrated RB model are listed in Chapter 3, Table 3.3-6 and Table 3.3-7.

#### **9B.6.1.2.1 Steel Bricks™ Equivalent Properties**

Geometric details of a typical Steel Bricks™ module are illustrated in Figure 9B.6-3. A typical Steel Bricks™ module consists of individual U-shaped bricks. The modules are fabricated by first cutting steel plate to desired shape or section. Welding shear studs to the inside of the plates is followed. It is then folded in press to create an L-Shaped section. After that the two L sections are welded to form a U-shaped brick. Finally, multiple modules are stacked and welded to form wall or flooring members. The welding takes place externally with no need for operators to be inside the panels.

The lateral and vertical resisting system elements are modeled using elastic shell elements with an equivalent thickness, elastic modulus, and material density calibrated to match the Steel Bricks™ effective stiffness and mass properties developed using ANSI/AISC N690, Appendix N9. The effective axial stiffness of these elements is developed using the method outlined in AISC Engineering Journal Article (Reference 9B.4-47).

Effective material properties used for the operating conditions represent the Best Estimate (BE) structure stiffnesses. Effective material properties for the fully cracked condition are referred to as the Lower Bound (LB) structure stiffnesses. For operating conditions, the effective flexural stiffness of the composite Steel Bricks™ section combines both the steel and concrete stiffnesses. The fully cracked condition is simulated by ignoring the contribution of concrete infill flexural stiffness.

Wall curvatures are not considered in the mathematical formulations of Steel Bricks™ properties. The contribution of shear connectors and Steel Bricks™ diaphragms to flexural stiffness are also not considered. The effects of liners used for corrosion and shielding protections is not considered in the calculation of FE properties.

#### **9B.6.1.3 Steel-Plate Composite Connections**

Full-strength welded (rigid) and bolted (semi-rigid) connections are used to connect the Steel Bricks™ structural elements in the integrated RB. Semi-rigid connections are typically used below grade to release some of the thermal induced stresses from the SCCV. Above grade, all connections are full-strength welded connections.

The integrated RB model captures the semi-rigid connections with six linear springs, with three translational and rotational stiffnesses each, at each pair of near coincident nodes where the walls and floors meet. The stiffnesses of the semi-rigid connections are estimated based on the component based model approach explained in Article 6.3 of the Eurocode 03 Part 8 (Reference 9B.4-48) and in Reference 9B.4-49.

The current floor-to-wall semi-rigid connections provide some flexibility in the floor in-plane and axial direction degrees of freedom, and in the floor out-of-plane bending. Similarly, wall-to-wall semi-rigid connections provide some flexibility in the wall axial and in-plane degrees of freedom, and in the wall out-of-plane bending. Each degree of freedom of the spring element is assigned either a semi-rigid stiffness or a rigid stiffness consistent with the orientation of release.

The integrated RB model captures the full-strength welded connections (rigid connections) with shared nodes or with a pair of rigid linear spring elements (translational and torsional) to restrain all six degrees of freedom. These rigid linear springs are also used at the intersection corners of the below grade RB floors and walls.

Figure 9B.6-4 shows the semi-rigid connection locations in the integrated RB model.

#### **9B.6.1.4 Reactor Pressure Vessel and CEPSS Connections**

The BWRX-300 RPV is supported vertically and laterally through the RPV skirt and laterally only through the top and bottom stabilizers as shown in Figure 9B-1. The RPV steel skirt is designed to allow thermal expansion in radial direction reducing the loads on the RPV pedestal. The bottom stabilizers are also used as shown in Figure 9B-1 to provide a lateral support to the control rod drive housing (CRDH) located below the RPV. The RPV is free for radial movement at the stabilizers and CRDH supports. The skirt, bottom stabilizers, and CRDH lateral support transfer lateral load directly to the pedestal. The upper stabilizers transfer lateral load to the CEPSS. The CEPSS framing is vertically and laterally supported by the pedestal and tangentially restrained by the SCCV wall through shear lugs that allow the CEPSS framing to radially move for thermal.

The RPV LMS model was initially developed with a single horizontal spring in each direction at the stabilizers, skirt, and CRDH locations. During its incorporation into the integrated RB FE model, these single springs were adapted so that the RPV lateral load can be transferred to the pedestal or CEPSS framing shown in Figure 9B.6-1. The RPV piping stiffness is considered to be decoupled for the bounding analyses.

#### **9B.6.1.5 Static and Dynamic Mass Properties**

The integrated RB model is analysed for the design loads described in Chapter 3, Section 3.5. These loads are assigned to the integrated RB model as static and dynamic masses.

Dead loads include self-weight and supplemental weight added to represent equipment and non-structural (i.e., staircases, steel platforms, partition walls, radiation shielding plates) dead loads. Static dead load analysis is performed by applying a 1-g acceleration to the static SSI model including all dead load masses. The dead load model masses also contribute to the seismic inertial loads and remain in the integrated RB dynamic model. Fifty percent of the live load and twenty-five percent of snow loads are modeled as equivalent mass in the integrated RB dynamic model.

As stated in Chapter 3, Subsection 3.3.1.2.2, the inertia associated with the vertical hydrostatic and horizontal impulsive hydrodynamic effects of the fluids contained in various RB pools are included in the integrated RB FE model. Methodology used to account for these loads is discussed in Chapter 3, Subsection 3.3.1.2.2.

Hydrostatic and hydrodynamic loads act on the pool floors and walls. Hydrostatic loads on the pool floors are captured by modeling the fluid volume in each of the pools as a vertical only mass during 1-g static SSI analyses. Hydrostatic loads on the pool walls are analysed as a pressure load during the structural analyses as demonstrated in Appendix 9B, Subsection 9B.9.2.3.

The hydrostatic fluid volume masses also act as the seismic inertial mass in the vertical direction during seismic analysis. The hydrodynamic impulsive loads are captured during seismic response analysis by assigning masses to the pool walls consistent with the impulsive pressure distribution. Hydrodynamic convective loads on the pool walls are analysed as a pressure load during structural analysis.

#### **9B.6.2 Seismic SSI Analysis Model**

Since the integrated RB is deeply embedded, the interaction of the structure with the surrounding subgrade is important for its structural integrity and its response under static and dynamic loads. In accordance with the guidance of NEDO-33914, Section 5.1, the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4, is implemented for the design of the BWRX-300 integrated RB structures to adequately account for the SSI effects requiring the development of a model that includes FE representations of the integrated RB (discussed in Subsection 9B.6.1) and the surrounding subgrade. To account for the SSSI effects with the surrounding Power Block

structures as discussed in Chapter 3, Subsection 3.3.1.2.1, coarse dynamic FE models of the RWB, CB, TB, and Reactor Auxiliary Bay structures, including foundations are added to the SSI model. These coarse mesh models represent the overall stiffness and mass distribution necessary to capture the global dynamic behaviour of the RWB, CB, TB and Reactor Auxiliary Bay structures. The models are inserted into the integrated model with the centreline of their mat foundations set a grade elevation (Level 0 m) and are separated from the RB model by a minimum gap of 760 mm in line with the discussion in Chapter 3, Subsection 3.3.1.2.8.

Figure 9B.6-5 shows an overview of the FE model developed for the seismic SSI analysis of the integrated RB structures. This model is assigned BE structure stiffnesses with static and dynamic masses and is referred to as the dynamic SSI model. The model is developed in ANSYS and then translated into ACS SASSI (see Chapter 3, Appendix 3B) format for the SSI analyses.

As described in Subsection 9B.3.3, the foundation supporting the turbine generator is structurally isolated from both the turbine shielding walls and shell structure for vibration control. Figure 9B.6-6 shows the LMS model used to represent the turbine generator pedestal structure and the isolated FE model used to model its isolated foundation in the seismic SSI analysis model.

#### **9B.6.2.1 Modeling of Soil-Structure Interfaces**

The soil and soil-structure interfaces of the seismic SSI analysis model consists of the following:

- Free-field subgrade model representing the properties of the subgrade materials that are approximated as continuum infinite horizontal layers resting on surface of uniform half-space
- Near-field subgrade volume representing the properties of subgrade materials below the RB and adjacent Power Block foundations and around the RB below grade exterior wall
- Excavated soil volume representing the properties of the volume of far-field subgrade replaced by the RB and adjacent near-field volume elements, and
- Contact spring elements connecting the near-field subgrade volume and the structures at interface nodes

##### **9B.6.2.1.1 Free-field Subgrade**

The free-field subgrade is represented in ACS SASSI by half-space continuum of horizontally infinite layers with uniform linear elastic properties resting on uniform elastic half-space. A viscous lower boundary of the subgrade model is established at depth that is at least three times the largest foundation dimension from the bottom of the foundation as discussed in Chapter 3, Subsection 3.3.1.2.2.

##### **9B.6.2.1.2 Near-field Subgrade Volume**

A single layer of near-field solid elements is placed below the BWRX-300 Power Block foundations and around the below grade portion of the RB exterior wall representing the properties of the near-field subgrade materials as shown in Figure 9B.6-5.

At the excavated volume perimeter, the FE of excavated soil volume (see Subsection 9B.6.2.1.3) and the near-field volume share the same nodes. The mesh of these two subgrade volumes aligns with every other mesh element of the RB exterior wall and mat foundation. Having the FE mesh of the near-field volume and excavated volume being twice the size of the refined one-step approach RB structural model helps reduce the ACS SASSI dynamic soil impedance calculations to practically reasonable durations. The mesh of the near-field volume aligns with the mesh of the surface mounted Power Block foundations.

Per guidance of NEDO-33914, Section 5.5, the BWRX-300 design conservatively does not consider the resistance of the temporary construction elements such as the excavation support structure and the lean concrete material used to fill the gap between the excavation edge and the RB exterior wall. As a result, the near-field soil volume in the FE model used for the design basis seismic SSI analysis is assigned the same properties as the far-field soil.

#### **9B.6.2.1.3 Far-Field Subgrade Excavated Volume**

The excavated volume model (see Figure 9B.6-7) consists of 3-D solid FEs representing the part of the horizontally layered subgrade displaced by the near-field volume and the embedded portion of the RB structure. The mesh of the excavated subgrade volume aligns with the layering of the far-field subgrade profiles and the near-field subgrade volume shown. The meshes of the excavated and near-field volumes are aligned at the perimeter, where the excavated volume and near-field volume solid elements share the same nodes.

#### **9B.6.2.1.4 Near-Field-Structure Interfaces**

Three contact linear spring elements are assigned at all near coincident nodes between the near-field volume and the FE models of the RB and adjacent Power Block structures as well at the nodes of the RB exterior wall to represent normal (radial) and friction (vertical and tangential) interaction mechanisms. Contact spring elements are assigned at near coincident nodes between the near-field volume mesh and every other node on the RB exterior wall and mat foundation mesh.

For seismic analysis, all contact springs in the dynamic SSI model are assigned high stiffnesses to simulate fully bonded conditions between the near-field volume solid elements and the structural shell elements to emphasize the effects of SSI and SSSI on the seismic response of the deeply embedded RB.

### **9B.6.3 1-g Static SSI Analysis Model**

The 1-g static SSI analysis provides design demands for gravity inertia loads and static earth pressure loads including surcharge loads from the foundations of adjacent Power Block buildings.

Two sets of 1-g static SSI analysis are performed to evaluate the unfactored and factored gravity demands on the integrated RB (see Table 9B.6-1) requiring the development of two 1-g static SSI analysis models.

The 1-g static SSI models have the same configuration and FE attributes as the dynamic SSI model developed for the seismic analysis described in Subsection 9B.6.2. The structural members of the 1-g static SSI models are assigned the same BE effective stiffness properties as the dynamic model. Differences between the static and dynamic models consist of the following adjustments made to the unit weight properties of shell and lumped mass elements of the 1-g SSI models:

- The removal of the live and snow mass properties from the 1-g static SSI Models. The fluid horizontal dynamic masses are not removed from the models since these are not activated in 1-g gravity analysis.
- The RPV total mass is increased to reflect the bounding refueling condition.
- The total mass of adjacent Power Block structures is increased by 10% in the unfactored and factored 1-g SSI analysis models for added safety margin. The 10% increase in the factored model is increased by an additional 33% due to factoring of loads for a total mass increase of 46%.

Per recommendations of NEDO-33914, Section 5.1.2, stiffness properties assigned to the contact springs in the 1-g static SSI models provide SSI interface conditions that emphasize the lateral earth pressure loads on the RB exterior wall below grade. A high stiffness is assigned to the contact spring elements in the direction normal to the RB exterior wall and mat foundation to simulate bonded conditions. The friction at the RB exterior wall and mat foundation is not considered by assigning very low stiffness properties to the contact springs in the vertical and tangential direction. Similarly, contact springs at the bottom of the surface mounted foundations of the surrounding Power Block structures are assigned a high stiffness in the vertical direction and low stiffness in the horizontal direction to maximize the effects of the surcharge loads on the lateral earth pressures on the RB exterior wall.

#### **9B.6.4 Static and Quasi-Static Analysis Models**

The static and quasi-static analyses provide design demands for static normal operating and accidental conditions loads, excluding thermal loads, not captured in the 1-g static SSI analysis. For more information on load cases considered in the analyses, refer to Chapter 3, Subsection 3.5.1.1.3 and Table 9B.6-1.

Static and quasi-static analyses are performed on standalone models of the integrated RB with either BE stiffness properties representative of low stress response conditions or LB structure stiffness properties representative of fully cracked concrete conditions when the structural members experience high stresses.

Static and Quasi-static analyses are performed using the subgrade stiffness sub-structuring method as discussed in Chapter 3, Subsection 3.5.1.1.3 in which the subgrade represented by solid FEs (see Figure 9B.6-8) is condensed into a single matrix element, called a super-element. This super-element is shown in Figure 9B.6-9 and represents the SSI impedance at nodes coincident with every other node of the RB exterior wall and mat foundation. For static and quasi-static analyses, the super-element is assigned LB subgrade stiffness properties to emphasize the deformations at the subgrade-structure interfaces resulting in conservative estimate of the member force and moment demands on the RB integrated structure.

#### **9B.6.5 Thermal Stress Analysis Model**

Thermal analyses provide stress demands due to normal operating and accidental load conditions as indicated in Table 9B.6-1.

Thermal stress analyses are performed on standalone models of the integrated RB that have the same configuration and FE attributes as the models developed for static and quasi-static analyses discussed in Subsection 9B.6.4. The standalone integrated RB model used for normal operating thermal analysis is developed with BE structure stiffness properties. The standalone integrated RB model used for the accidental thermal analysis is developed with LB structure stiffness properties assigned to the exterior pool walls, SCCV wall, inner mat foundation and SCCV top slab, and BE structure stiffness properties elsewhere.

Thermal stress analyses use super-elements assigned with upper bound subgrade stiffness properties to emphasize the constraints at the subgrade-structure interfaces against the thermal expansion resulting in upper bound estimates of thermal stresses in the RB structures.

#### **9B.6.6 Validation of FE Models**

##### **9B.6.6.1 Validation of Integrated Reactor Building Model Translation into ACS SASSI Format**

The validation of the translation of the integrated RB dynamic model into the ACS SASSI format is performed by comparing the results of:



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

- A fixed base eigenvalue analyses of the dynamic model in the ANSYS format, and
- A SASSI analysis of the integrated RB dynamic model resting on the surface of a very stiff half-space simulating fixed base conditions.

Eigenvalue analyses are performed in ANSYS using the Block Lanczos method for the first 500 modes up to 35.2Hz, where a minimum of 85% cumulative mass participation ratio is reached (See Figure 9B.6-10). The integrated RB significant mode frequencies and mass participation ratios in each orthogonal direction are as follows:

- Primary X mode occurs in the RB, SCCV and CEPSS at a frequency of 2.25 Hz with a 58% mass participation.
- Primary Y mode occurs in the RB, SCCV and CEPSS at a frequency of 2.30 Hz with a 41% mass participation.
- Primary Z (vertical) mode occurs in the RB, SCCV and pedestal at a frequency of 7.73 Hz with a 57% mass participation.

ACS SASSI analyses of “fixed base” surface mounted RB dynamic model is performed from 0.1 to 70Hz and Acceleration Transfer Functions (ATFs) amplitudes for the responses in the three orthogonal directions at key nodal locations plotted. Figure 9B.6-10 compares the normalized transfer functions at the integrated RB roof node #129688 (see Appendix 9B.B, Figure 9B.7-3) with fixed base eigenvalue cumulative mass ratio plots for each of the three orthogonal directions. As shown in Figure 9B.6-10, the first peaks of the transfer functions occur at 2.2Hz and 6Hz with the second peaks occurring at 7.3Hz and 7.7Hz in the horizontal and vertical directions, respectively. The fixed base integrated RB structure transfer function peaks are consistent with the primary modes of the eigen value analysis.

#### **9B.6.6.2 Validation of ACS SASSI Stress Results**

The accuracy of stress results obtained from the ACS SASSI translated model are validated by comparing element stress responses obtained from the two following analyses of the integrated RB model:

- ANSYS static analysis of a fixed base model with 1-g gravity loads applied in the three orthogonal directions and
- ACS SASSI quasi-dynamic analysis of the model resting on very stiff half-space with 1-g acceleration applied in each of the three orthogonal directions.

Locations selected for comparison of stress results include the SCCV top slab (Elevation 4.9 m), northern and southern wing walls below grade, RB exterior wall below grade, SCCV wall and RPV pedestal. For each location, the critical load path is selected for each direction of loading and then compared side by side as shown in Figure 9B.6-11 for the SCCV top slab.

As demonstrated in Figure 9B.6-11, all side-by-side comparisons performed for each of the critical load paths are consistent between ANSYS and SASSI in terms of magnitude and distribution of demands. This confirms the translation of the integrated RB dynamic model was properly performed.

#### **9B.6.6.3 Validation of Incorporation of RPV Lumped Mass Stick Model**

The incorporation of the RPV LMS model with the supporting integrated RB model is validated by comparing the dynamic characteristics of the RPV LMS model within the integrated RB model with benchmark results the eigenvalue analysis of the standalone RPV LMS model.

Table 9B.6-2 reports the fundamental modes of the RPV LMS model and pairs each mode with its corresponding mode observed in the modal analysis described in Section 9B.6.6.1. As shown in Table 9B.6-2, the modal frequencies from these two sources are within the 10% variation requirement in Section 5.3 of NEDO-33914.

#### **9B.6.6.4 Validation of Dynamic Models of Adjacent Power Block Structures**

The coarse FE models of the adjacent Power Block structure that are coupled with the integrated RB are required to represent their global dynamic properties for use in seismic SSI analyses. Consequently, validation analyses are performed to demonstrate that these dynamic models can capture all important global modes of vibration of the Power Block structures by comparing results from:

- Fixed base eigenvalue analyses of refined design basis dynamic FE models of the adjacent Power Block structures, and
- SASSI analyses of the coarse dynamic FE models resting on the surface of a very stiff half-space simulating fixed base conditions.

Validation results demonstrate that the dynamic characteristics of the coarse dynamic models match reasonably well in total mass and dominant dynamic response with those from the refined models as demonstrated in Figure 9B.6-12, Figure 9B.6-13, Figure 9B.6-14, Figure 9B.6-15 and Figure 9B.6-16. Therefore, the coarse FE models of the adjacent Power Block structures provide an adequate representation of the global dynamic properties for the SSI analyses of the integrated RB.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B.6-1: Integrated RB Analyses Cases and Model Requirements**

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Seismic Inertial, earth pressure, hydrodynamic: impulsive pressure on pool walls and slabs	E	Seismic SSI (ACS SASSI)	Effective BE	BE strain compatible LB strain compatible UB strain compatible	Rigid in all three directions	Rigid in all three directions	Coupled dynamic models of RB and adjacent Power Block structures
Unfactored dead load, vertical hydrostatic load, and static earth pressure	D, F, H	1-g static SSI (ACS SASSI)	Effective BE	Static	Rigid in normal direction. Soft in the other 2 direction	Rigid in vertical direction. Soft in the horizontal directions	Coupled static models of the RB and adjacent Power Block structures
Factored dead load, vertical hydrostatic load and static earth pressure	1.2D, 1.2F, 1.6 H			UB unit weight			
				UB Poisson's ratio			
Live load on floors	L	Static subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	LB static Massless LB Poisson's ratio LB Young's modulus	Rigid in normal direction. Soft in the other 2 directions	Rigid in vertical direction only	Standalone RB model
Critical crane position reaction loads	C						
Hydrostatic and hydrodynamic breathing mode pressures on pool walls	F or E						
Wind load on above grade RB exterior wall and roof	W						
Tornado wind load above grade RB exterior wall and roof	W <sub>t</sub>						
Post-accident internal flooding loads on walls and mat foundation for maximum water level	H <sub>a</sub>						
Normal operating and SIT uniform containment pressure	P <sub>0</sub> , P <sub>t</sub>						
Accidental pressure loads on RB sub-compartments	P <sub>a</sub>						
Accidental uniform containment pressure	P <sub>a</sub>						

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
			effective BE elsewhere				
Seismic hydrodynamic sloshing load on pool walls and accidental seismic torsion	E	Static subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	LB static Massless LB Poisson's ratio LB Young's modulus	Soft in all three directions	Rigid in all three directions	Standalone RB model
Normal snow/rain on roof	S, R						
Extreme Precipitation (Snow) pressure on roof	S <sub>x</sub>						
Groundwater hydrostatic pressure on mat foundation and RB exterior wall below groundwater level	H					Rigid in vertical direction only except at mat foundation centre rigid in all three directions	
Normal operating and DBA nozzle, equipment and piping reaction loads	R <sub>o</sub> , R <sub>a</sub>						
Local loads on containment during DBA	R <sub>r</sub> , R <sub>b</sub>						
Additional lateral rock pressure on RB exterior wall	H, E						
Normal operating thermal loads	T <sub>0</sub>	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	UB static Massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
Accidental thermal loads on RB sub-compartments	T <sub>a</sub>		LB for cracked Steel Bricks™ elements and effective BE elsewhere				
Accidental thermal loads on containment	T <sub>a</sub>		LB for cracked Steel Bricks™ elements and effective BE elsewhere				

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B.6-2: Validation of RPV Lumped Mass Stick Model**

RB Dynamic Integrated Model Modes			LMSM Modes		Description of Mode
Mode	Frequency	Direction	Mode	Frequency	
2	2.29	Y	1	2.28	1st natural frequency for steam separators & standpipes modes.
3	2.29	X	2	2.28	
5	3.82	X	3	3.82	fuel modes
6	3.82	Y	4	3.82	fuel modes
21	8.86	Y	5	9.02	Mixed mode of Separators & Standpipes, fuel, plus 1st natural frequency for chimney & Shroud
22	8.89	X	6	9.02	
28, 32	9.29, 9.64	Y	7	9.44	Mixed mode of 1st natural frequency for control rod guide tube, chimney & shroud, separator & standpipes
29, 33	9.3, 9.68	X	8	9.44	
56	12.97	XY	9	12.85	Mixed mode of separator & standpipes and fuel
57	12.97	XY	10	12.85	
62	13.5	XY	13	13.84	1st natural frequency for CRDH first mode
63	13.5	XY	14	13.84	1st natural frequency for CRDH first mode
65	13.7	XY	11	13.64	Mixed mode of CRDH, Separator and fuel
66	13.7	XY	12	13.64	Mixed mode of CRDH, Separator and fuel

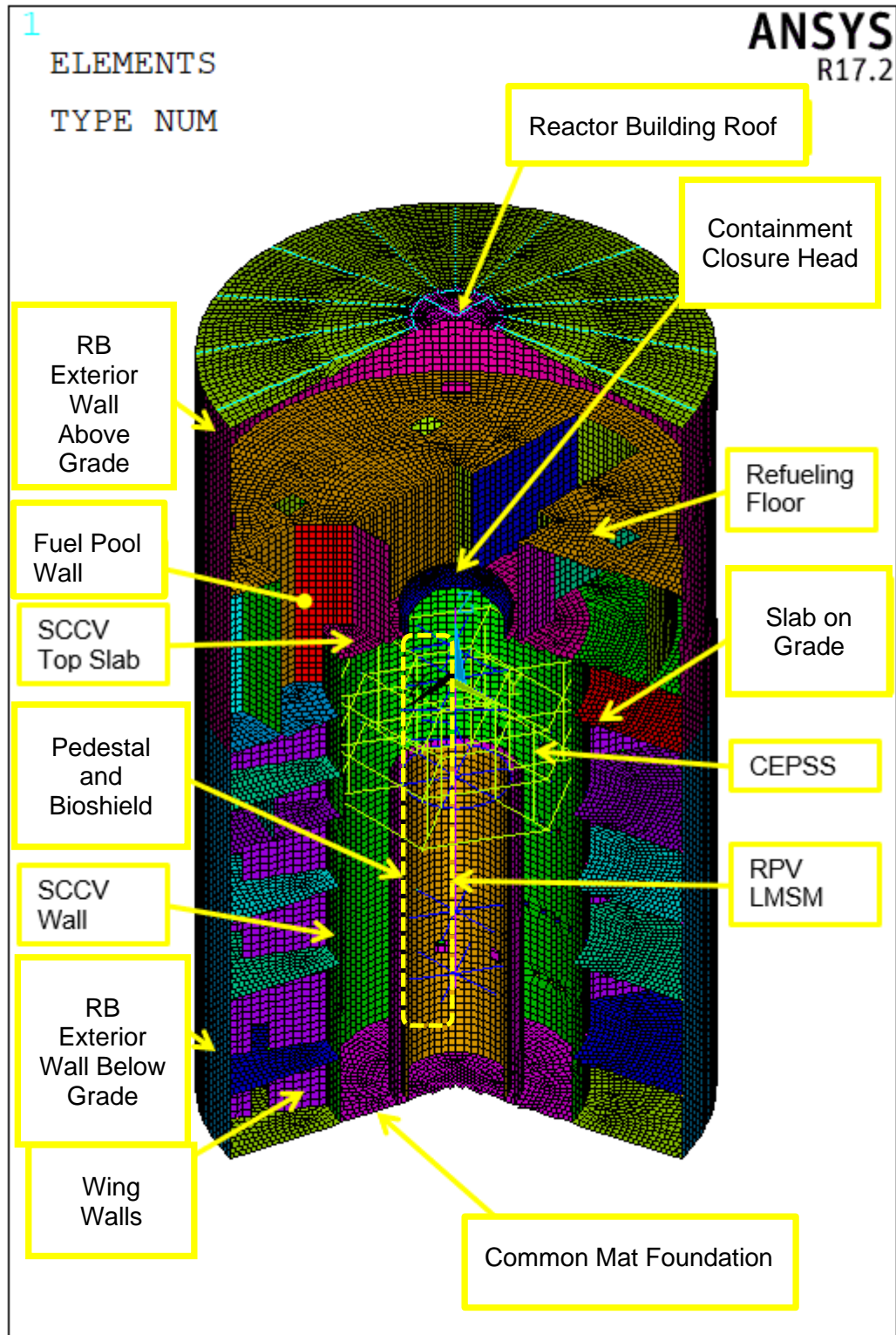
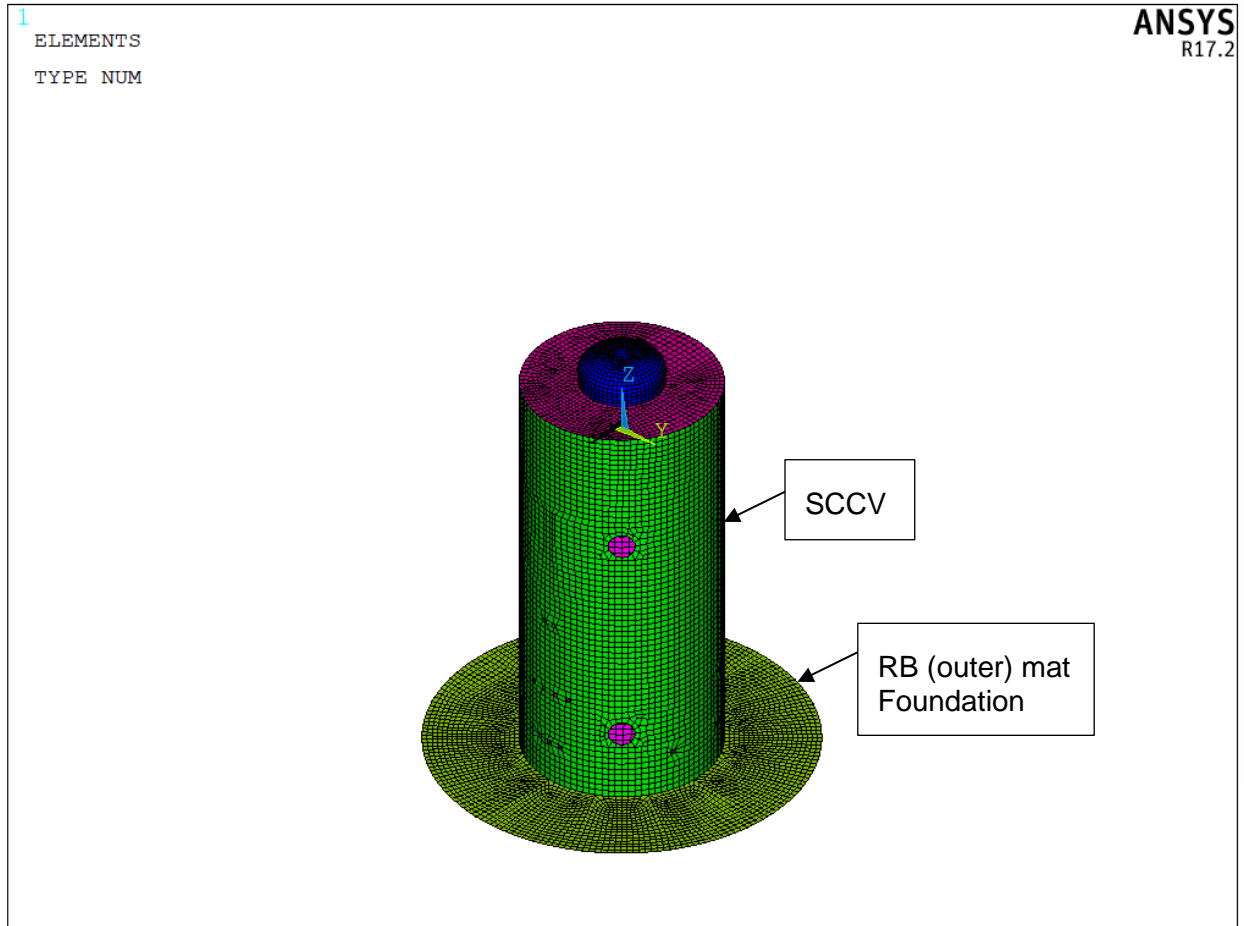
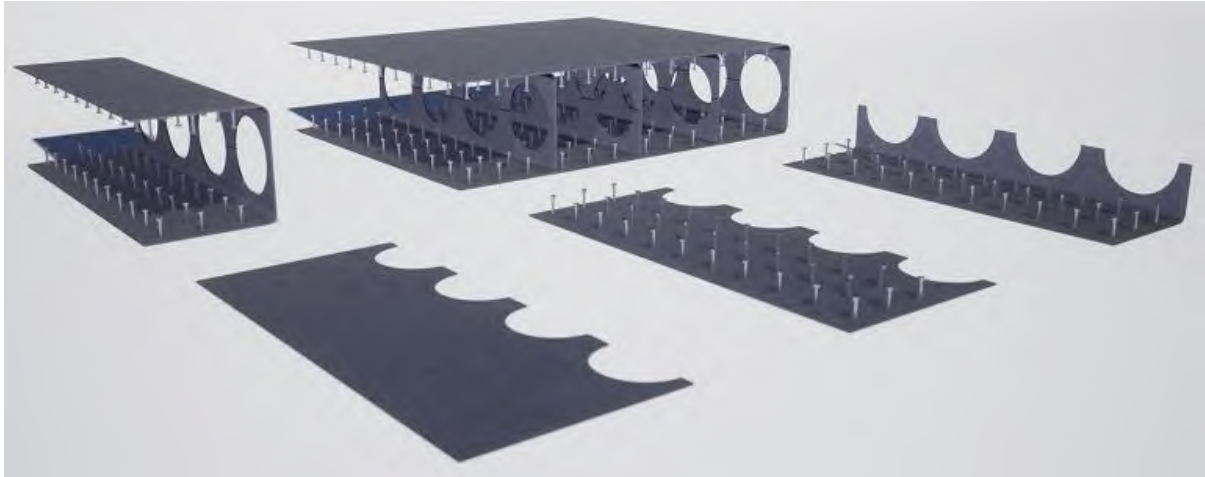


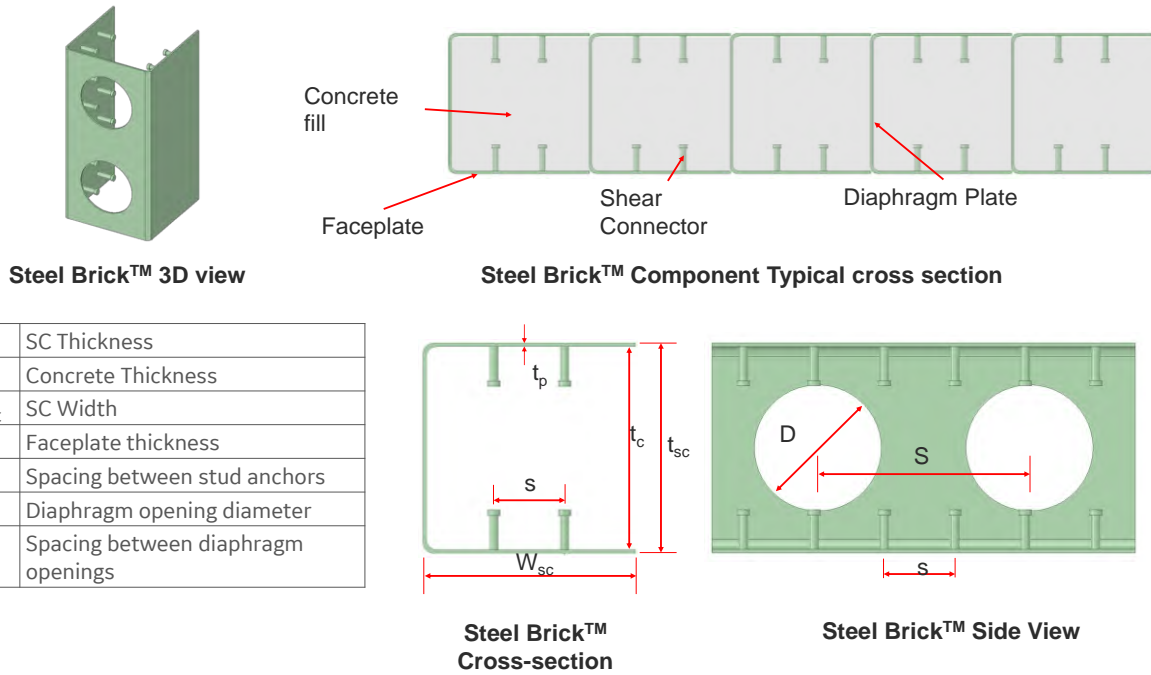
Figure 9B.6-1: Overview of Integrated Reactor Building Finite Element Model



**Figure 9B.6-2: Steel-Plate Composite Containment Vessel Structural Model with Penetrations**



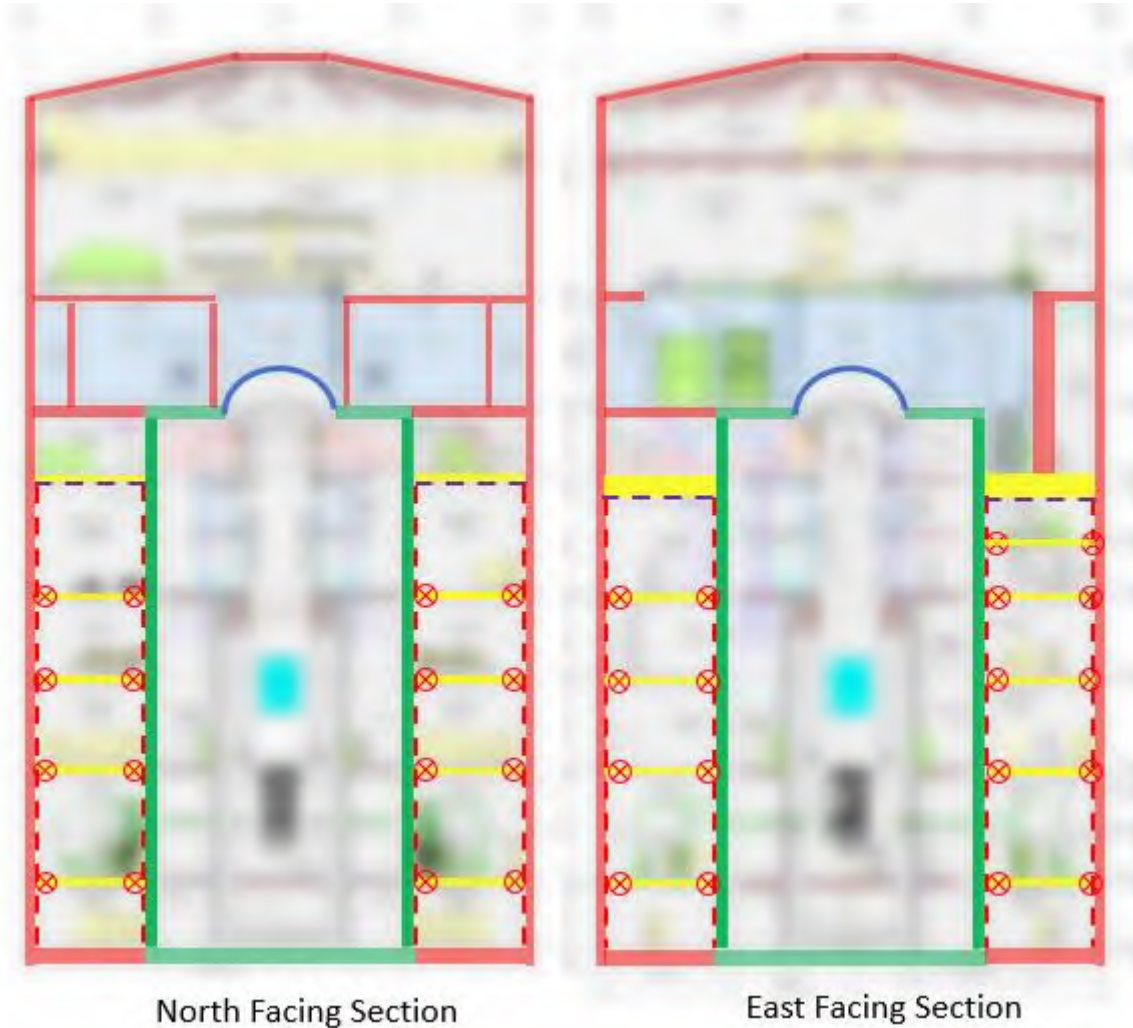
(a) Steel Bricks™ Fabrication Steps




(b) Steel Bricks™ Geometrical Details

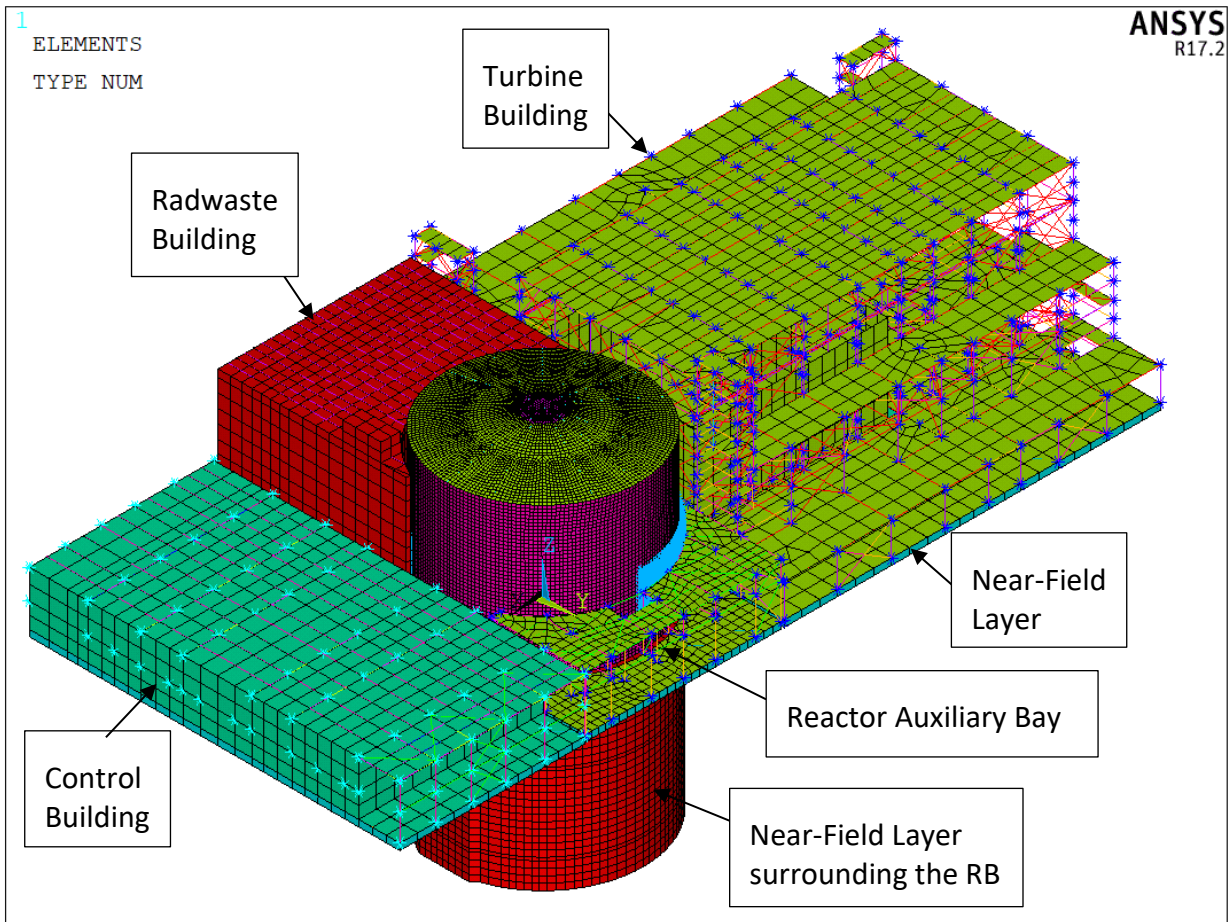
Figure 9B.6-3: Typical Steel Bricks™ Section





-  Semi-rigid connections between Steel Bricks™ floors and Walls
- Semi-rigid connections between wing walls and RB and SCCV shaft walls
- Semi-rigid connections between wing walls and Steel Bricks™ Floor at Level 0 m

**Figure 9B.6-4: Semi-Rigid Connection Locations**



**Figure 9B.6-5: Integrated Reactor Building Seismic SSI Analysis Model (Excluding Free-Field Subgrade Model)**

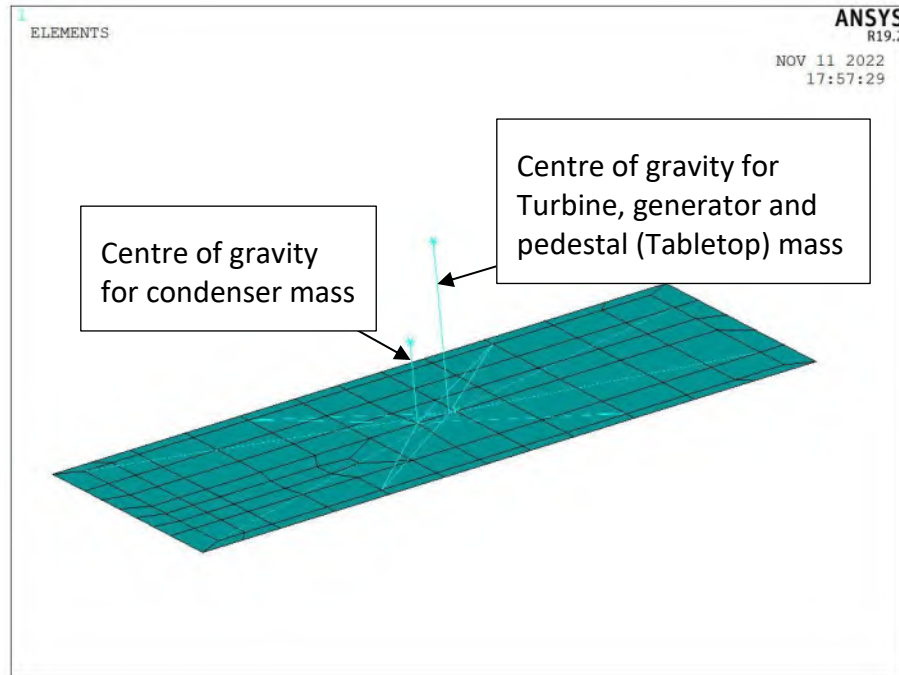


Figure 9B.6-6: Turbine Generator Pedestal and Isolated Foundation at Grade

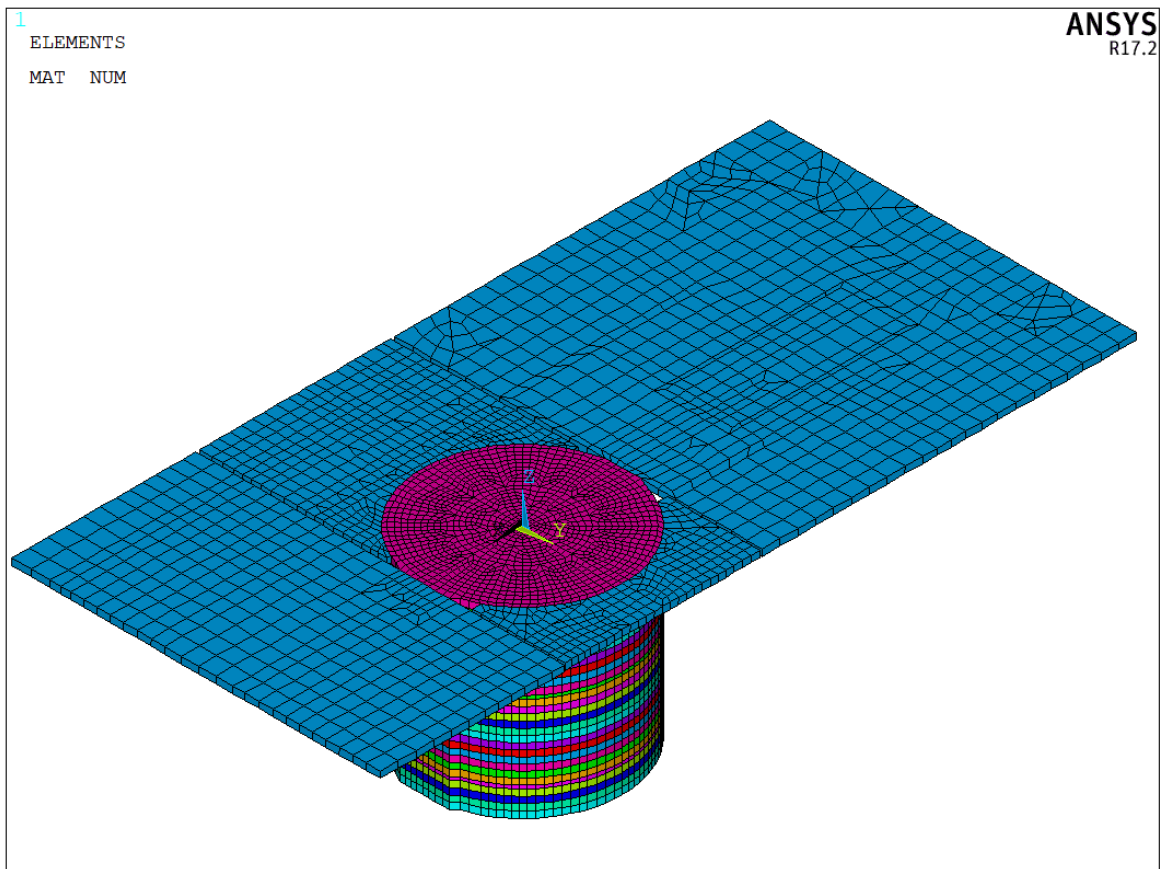
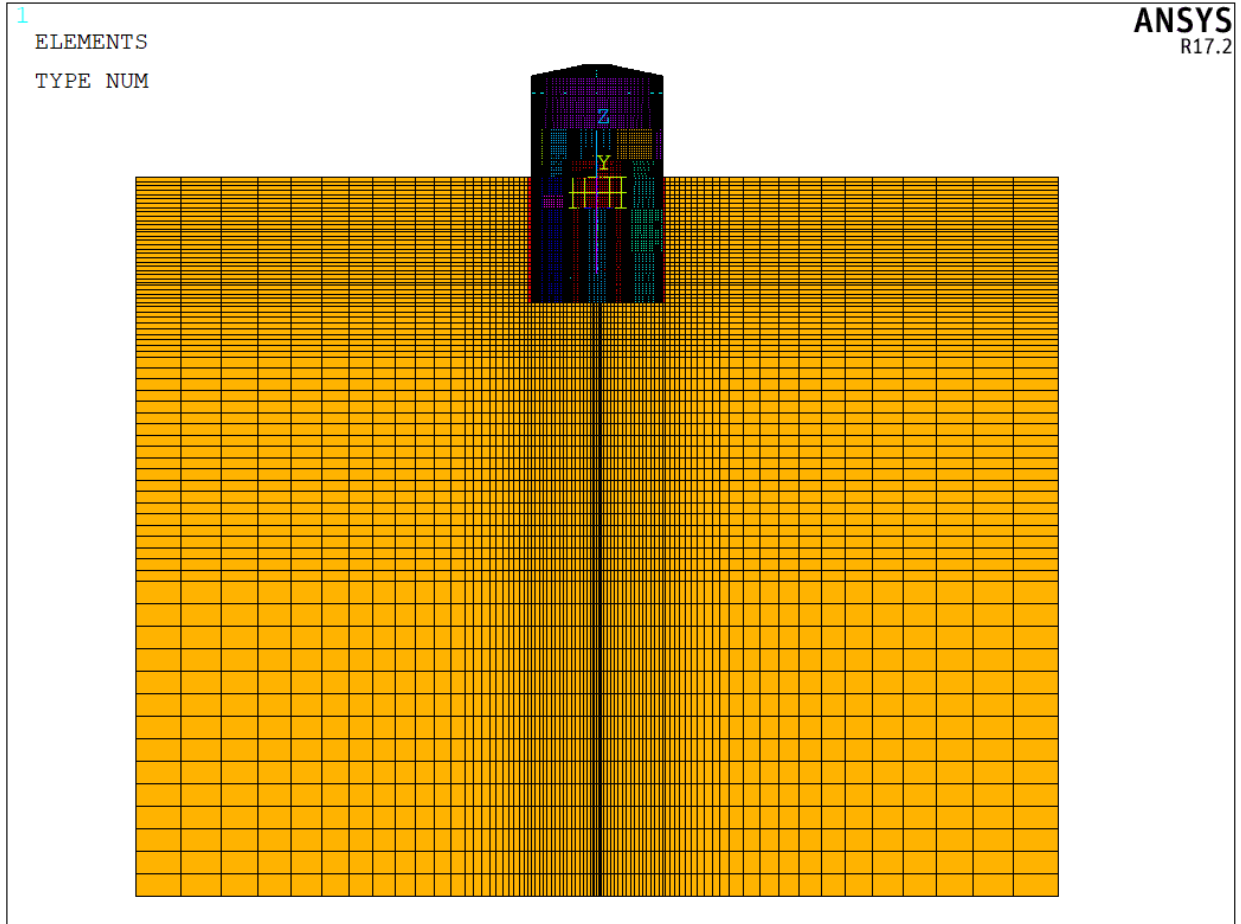
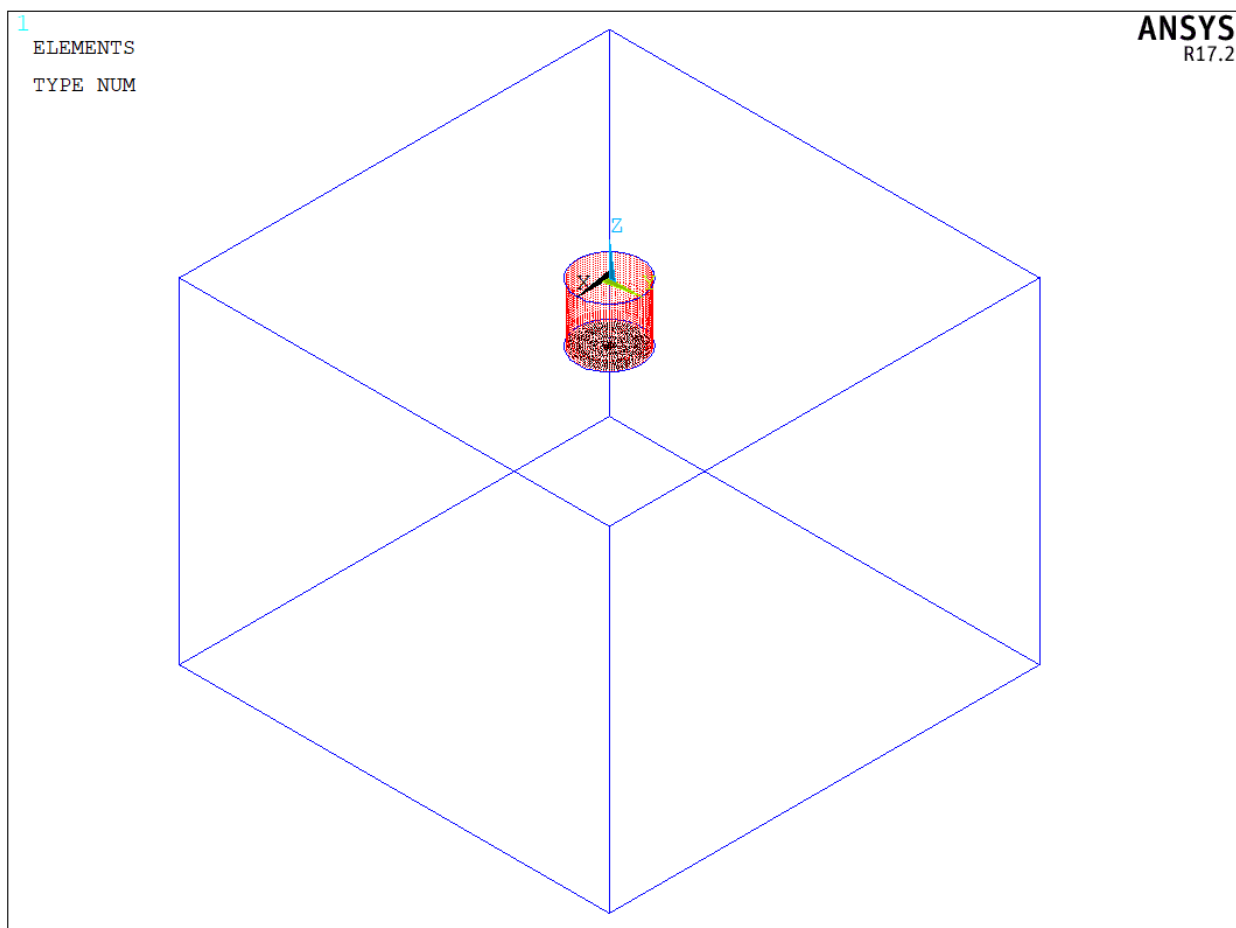


Figure 9B.6-7: Excavated Subgrade Volume Element Model

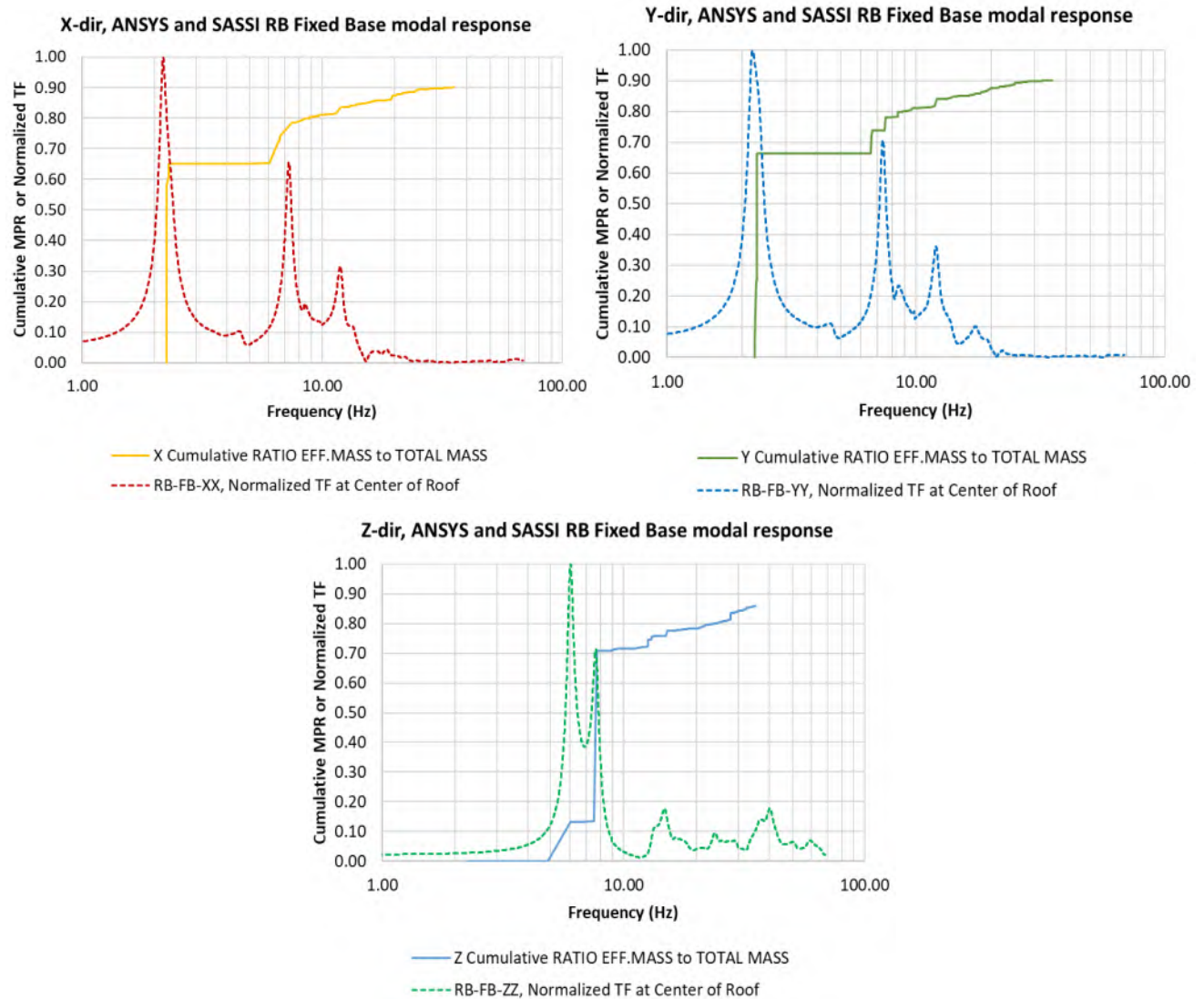


**Figure 9B.6-8: Finite Element Model for Calculation of Subgrade Impedance**



**Figure 9B.6-9: Super-Element and Reactor Building Contact Springs**

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



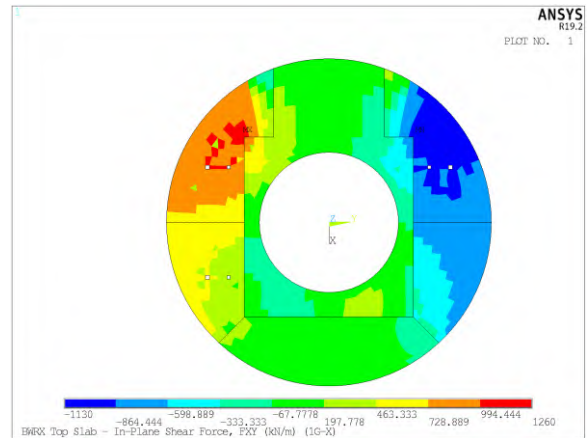
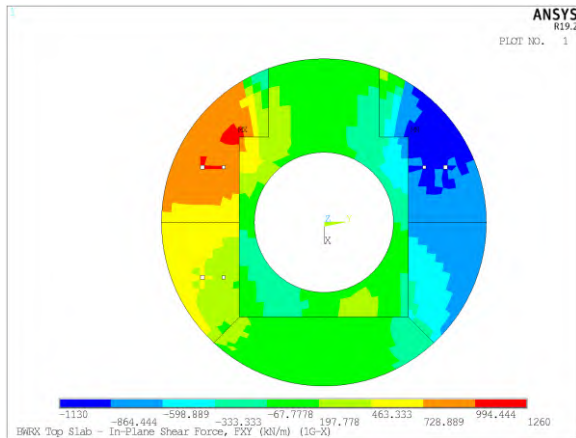
**Figure 9B.6-10: Dynamic Comparison at Centre of Reactor Building Roof in Three Orthogonal Directions**



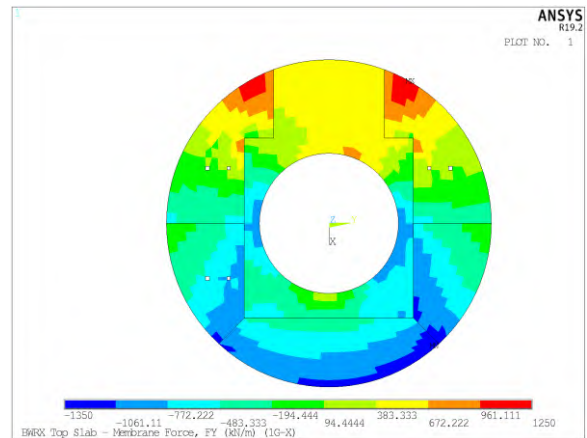
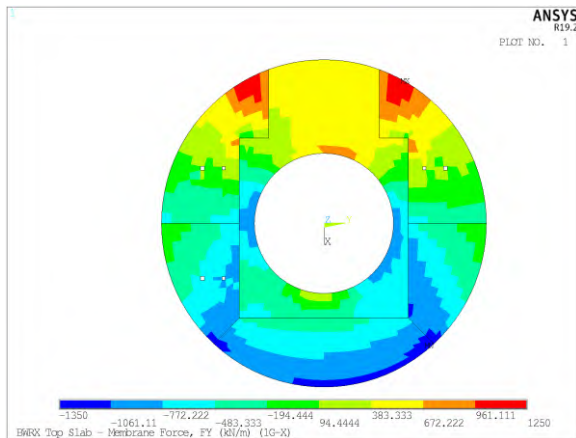
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

ANSYS

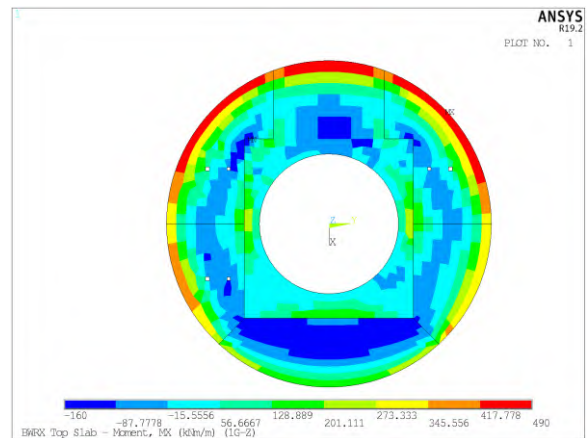
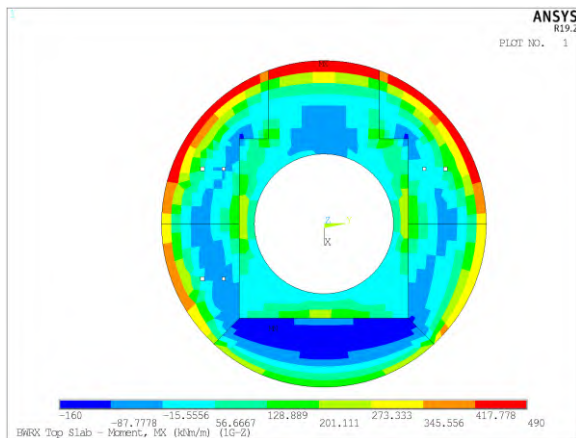
ACS SASSI



Top Slab 1Gx, In-plane shear



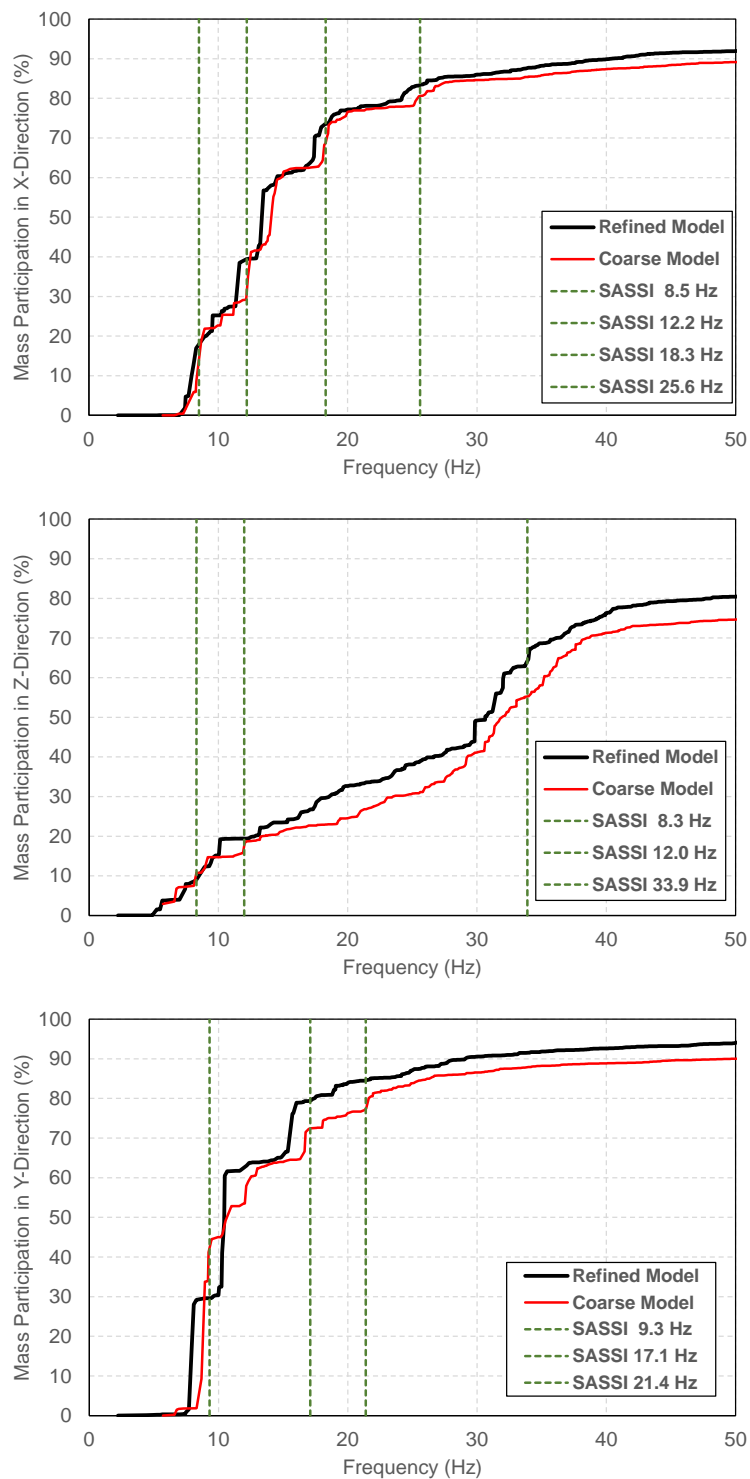
Top Slab 1Gx, Tangential membrane force



Top Slab 1Gz, Out-of-plane bending moments

Figure 9B.6-11: SCCV Top Slab Comparison

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.6-12: Cumulative Mass Participation Comparison for the Radwaste Building Refined and Coarse Models**



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

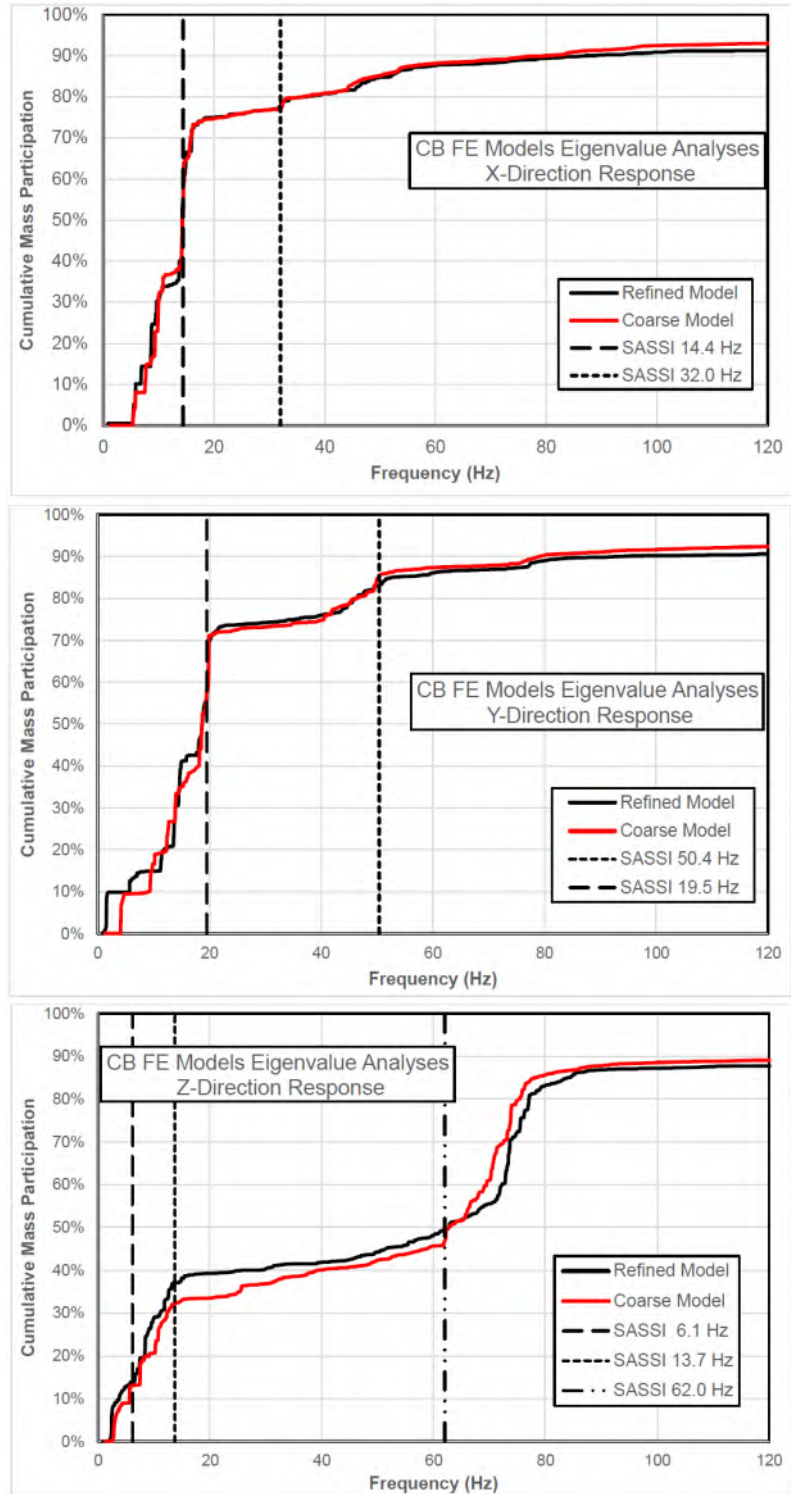
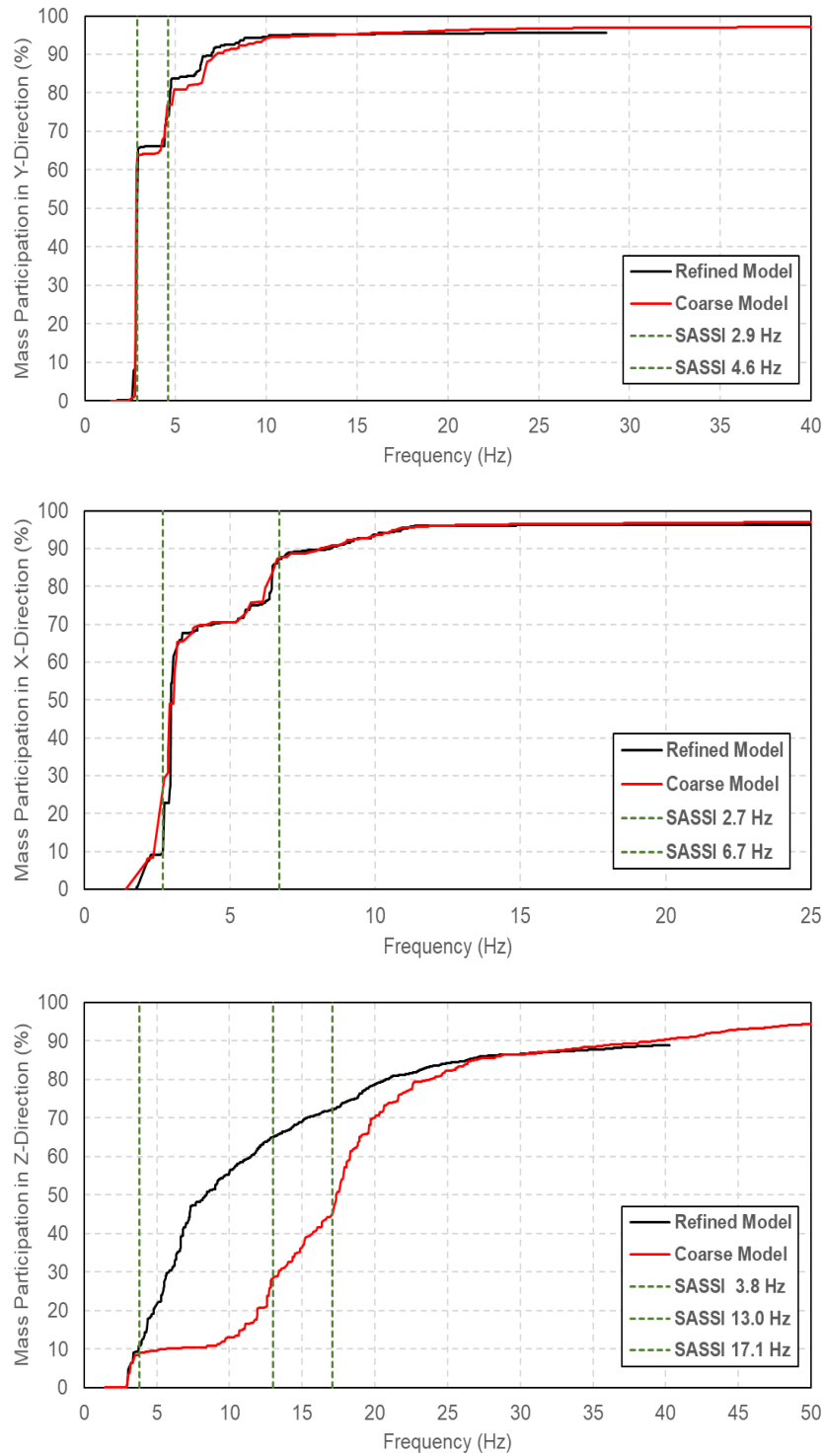


Figure 9B.6-13: Cumulative Mass Participation Comparison for the Control Building Refined and Coarse Models

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.6-14: Cumulative Mass Participation Comparison for the Turbine Building Steel Shell Refined and Coarse Models**

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

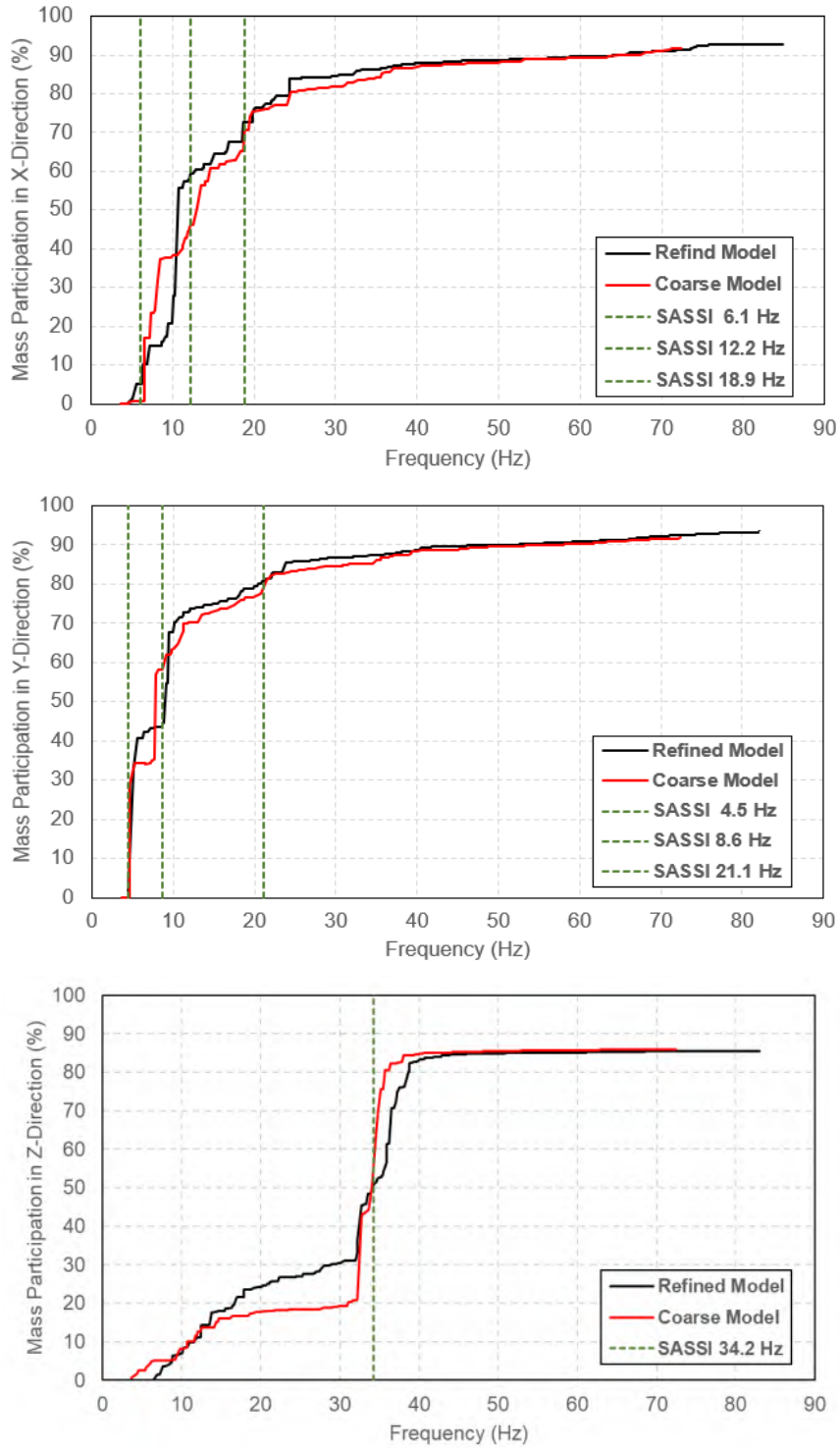
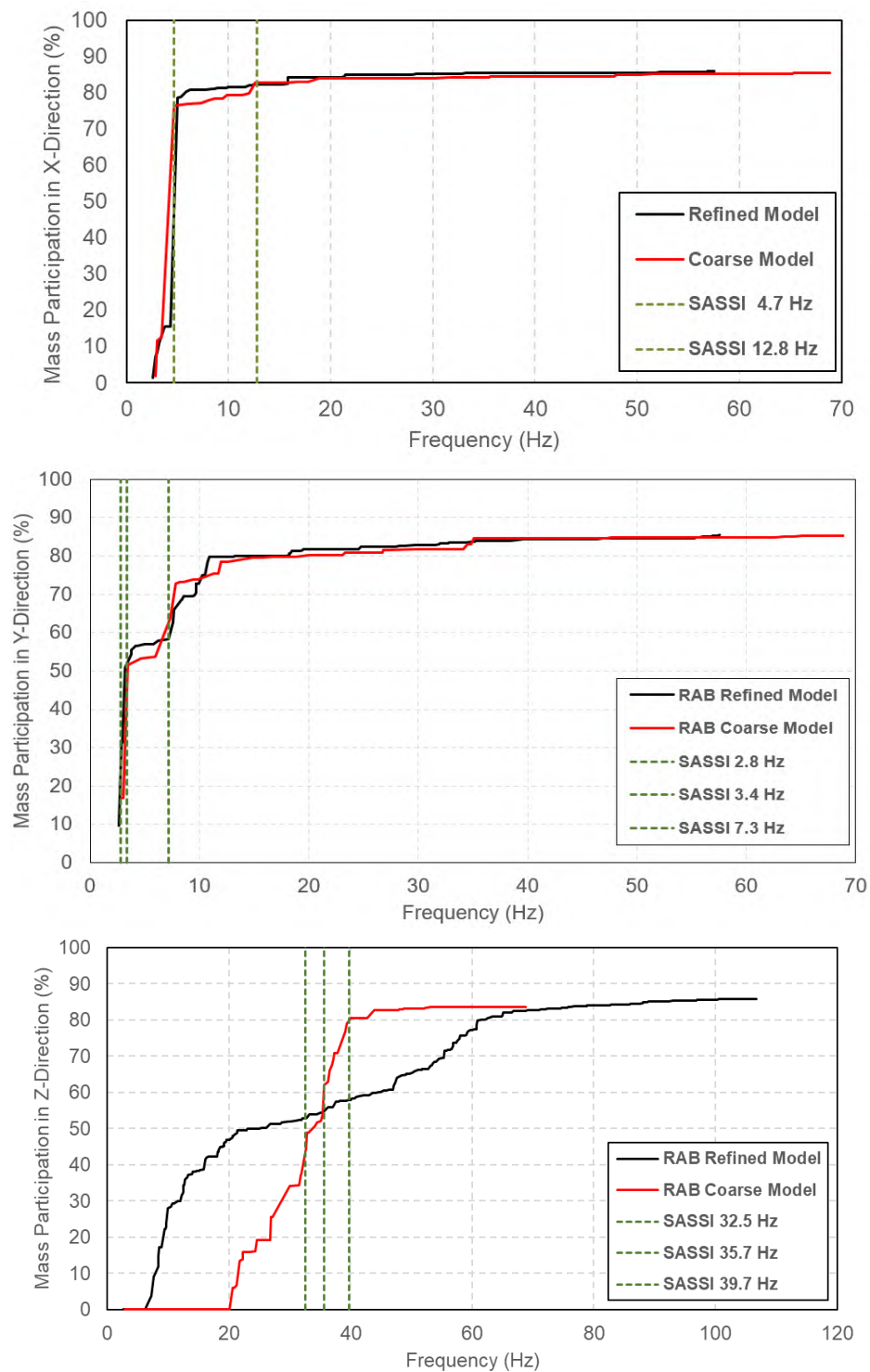


Figure 9B.6-15: Cumulative Mass Participation Comparison for the Shield Wall Area Structure in the Turbine Building

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.6-16: Cumulative Mass Participation Comparison for the Reactor Auxiliary Bay Refined and Coarse Models**

## **APPENDIX 9B.B – SEISMIC SOIL-STRUCTURE INTERACTION ANALYSIS**

### **9B.7 Introduction**

This appendix documents the results of the seismic SSI analysis of the integrated RB.

#### **9B.7.1 SSI Analysis Methodology**

Seismic demands for the design of the Seismic Category A and B SSC at the DNNP site are obtained from the seismic SSI analysis of the FE model discussed in Appendix 9B.A, Subsection 9B.6.2, evaluated for the site-specific bounding input ground motions and subgrade properties presented in Chapter 3, Subsection 3.3.1.1. These parameters are developed to conservatively cover the variabilities and address uncertainties in the geotechnical properties of the subsurface materials and seismicity of the DNNP site.

The seismic SSI analysis is performed using the one-step analysis approach discussed in Chapter 3, Subsection 3.3.1.2.1 which allows seismic stress demands to be obtained directly from the results of the SSI analysis. Frequencies selected for each of the subgrade profile cases in the seismic SSI analysis are provided in Subsection 9B.7.4.

##### **9B.7.1.1 Stress Calculation**

Shell element forces and moments obtained from the seismic SSI analysis are provided at the centre of the element with respect to the element local coordinate system. To calculate the co-directional stress demands, the time histories of responses due to the three earthquake components are combined algebraically for each element using the time-step-by-time-step approach as discussed in Chapter 3, Subsection 3.3.1.2.6. The combined stress demands are then transformed from the local coordinate system into a global cartesian or cylindrical coordinate system for use as input for the design of the structural members.

Average stress results from the five sets of Acceleration Time Histories (ATHs) used as input to the seismic SSI analysis represent the response of the structure for each SSI analysis case considered. Stress results obtained from the analyses of the three subgrade profile cases are enveloped to provide enveloping DNNP site-specific seismic stress demands that account for the uncertainties and variations of the subgrade properties.

##### **9B.7.1.2 In-Structure Response Spectra Calculation**

Per Chapter 3, Subsection 3.3.12.7, the In-Structure-Response Spectra (ISRS) are calculated at 301 frequency points equally distributed on the logarithmic scale at the frequency range from 0.1 Hz to 100 Hz to ensure the ISRS are sufficiently close to the peak response frequencies of the supporting structure. The ISRS are calculated at equipment locations for multiple damping ratios spanning the range corresponding to that of equipment under consideration.

The SSI analysis is performed separately for each one of the three directional components of input ground motion using each of the five sets of ATHs for each subgrade profile analysis case. For each set, the ISRS obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response using the Square-Root-of-the-Sum-of-Squares (SRSS) method. For each subgrade profile analysis, an average ISRS is computed from the five sets of ATHs to represent the amplitude and frequency content of the in-structure motion at a particular location as illustrated in Figure 9B.7-1. The ISRS from the bounding site-specific subgrade profile analysis cases are then enveloped as illustrated in Figure 9B.7-2. To address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology, peaks of the enveloping ISRS are broadened by  $\pm 25\%$  which exceeds the 15% requirement specified in Chapter 3, Subsection 3.3.1.2.7. Sharp valleys between peaks are also filled, as shown in Figure 9B.7-2, to account for the uncertainties in subgrade properties.

### **9B.7.1.3 Relative Displacement Calculation**

Relative displacements are calculated for key nodal locations throughout the RB relative to the free-field at mat foundation elevation. For each set of ATHs, the maximum relative displacement obtained from the analysis of each of the three ground motion components are combined using the SRSS method to get the total co-directional response. From the results of the analysis of each subgrade profile, average relative displacements from the five sets of ATHs are computed representing the maximum relative displacement for the considered subgrade profile. The relative displacements obtained from the analysis of the LB, BE and UB subgrade profile cases are enveloped to provide DNNP site-specific relative displacement demands that account for the uncertainties and variations of the subgrade properties.

### **9B.7.2 Key Nodal Locations**

As stated in Chapter 3, Subsection 3.3.1.2.5, responses at key nodal locations are used to demonstrate the accuracy of the seismic SSI analysis results, to determine analysis cutoff frequencies, and to evaluate the effects of SSI parameters variations based on comparisons of results from different seismic SSI analysis cases.

Key nodal locations selected per the criteria in Chapter 3, Subsection 3.3.1.2.5 for the verification and validation of the integrated RB seismic SSI analysis and results are provided in Table 9B.7-1 and illustrated in Figure 9B.7-3.

### **9B.7.3 Key Structural Members**

Seismic stress demands are calculated for key structural members and compared between SSI analysis cases to understand the effects of subgrade properties on the seismic response of the soil-structure system.

Table 9B.7-2 presents the key structural members selected based on their critical support function of safety-related equipment, critical importance within the lateral force resisting system, or critical location along the seismic load path. These key structural members are also illustrated in Figure 9B.7-3.

### **9B.7.4 Design Basis Seismic Analysis Cases and Selected Frequencies of Analysis**

Three seismic SSI analysis cases, representing the LB, BE and UB subgrade profile cases discussed in Chapter 3, Subsection 3.3.1.1.5, form the design basis for the integrated RB at the DNNP site.

The three seismic SSI analysis cases are performed for frequencies starting at 0.006 Hz to the respective cutoff frequency for each of the subgrade profile cases. Following the selection criteria in Chapter 3, Subsection 3.3.1.2.1, the following cutoff frequencies are adopted for the analysis:

- 22 Hz for the LB case
- 41 Hz for the BE case
- 68 Hz for the UB case

To evaluate the adequacy of these selected cutoff frequencies, the normalized ATFs and ISRS results from the analyses of the LB, BE and UB subgrade profiles are compared as explained in Subsections 9B.7.4.1 and 9B.7.5.2. The comparisons of the normalized ATHs and ISRS results in Subsections 9B.7.4.1 and 9B.7.5.2 show that the selected cutoff frequencies for the LB and BE subgrade profile cases of 22 Hz and 41 Hz are adequate for calculation of enveloping SSI responses. The SSI analysis of the UB subgrade profile provides bounding responses at frequencies significantly higher than the cutoff frequencies of the LB and BE subgrade profiles resulting in enveloping SSI results that bound the effects of variation of subgrade properties at the DNNP site.

The SASSI frequency domain analysis calculates the response of the system at selected “Structure” frequencies defined as the number of frequencies necessary to adequately characterize the response of the full SSI system. The results for the response at frequencies between the selected frequencies are interpolated. The ATF results for the responses at key nodal locations are plotted and inspected to ensure the accuracy of the transfer function interpolation as illustrated in Figure 9B.7-4 for the RB roof. The dots in the ATF plots in Figure 9B.7-4 represent the calculated “Structure” frequencies, and the lines show the interpolated ATFs. The smooth curves and the presence of calculated frequencies near peaks, troughs and inflection points indicate that the spacing of analysis frequencies is adequate for capturing the seismic response of the analysed soil-structure system over the frequency range of interest.

#### **9B.7.4.1 Subgrade Properties Variation Effects**

Figure 9B.7-5 compares the ATFs representing the co-directional responses at the edge of the RB roof in the three orthogonal directions obtained from the analysis of the LB, BE and UB subgrade profile cases. These ATFs are normalized with respect to the free-field response at surface by removing the large amplitude ATF peaks at the embedment soil column frequencies to better depict the peak responses of the RB structure.

The plots in Figure 9B.7-5 show distinct dominant frequencies for the SSI system depending on the subgrade profile analysis. Following the expected trend, ATF peak frequencies obtained from the LB profile analysis are associated with the lowest frequency, followed by higher ATF peak frequencies for the BE and UB cases as shown in Figure 9B.7-5 and shown in Table 9B.7-3. Frequencies for the horizontal direction of motion, associated with the shear-wave velocity for the subgrade, are also found to be smaller than those for the vertical direction, associated with the larger compression wave velocity for the subgrade.

As illustrated in Figure 9B.7-5, the amplitudes of the normalized ATFs obtained from the LB and BE subgrade profile analysis cases fall well below unity at frequencies that are smaller than their respective cutoff frequencies (22 Hz for LB and 41 Hz for BE), thus confirming the adequacy of the selected cutoff frequencies. The plots also show that the amplitudes of the normalized ATFs obtained from the UB subgrade profile analysis govern the seismic response at higher frequencies, and therefore capture the governing seismic demands up to its cutoff frequency of 68 Hz.

#### **9B.7.5 DNNP Seismic Response Characteristic**

##### **9B.7.5.1 Seismic Structural Stress Responses**

This section presents plots of main stress components in key structural members to depict the characteristics of how the RB integrated structures resist the DNNP site-specific seismic loads. Member force and moment results obtained from the analysis of the LB, BE and UB subgrade profile cases are compared to evaluate the effect of subgrade variations on the seismic stress responses of the RB integrated structures.

Figure 9B.7-6 through Figure 9B.7-17 compare the in-plane and out-of-plane membrane forces and bending moments on the above and below grade portions of the RB exterior wall obtained from the LB, BE and UB subgrade profiles. Due to the cylindrical shape of the wall, the out-of-plane dynamic pressure forces applied on the below grade portion of the RB exterior wall are primarily resisted by tangential (hoop) membrane forces as shown in Figure 9B.7-6 through Figure 9B.7-8. Comparing the forces due to the LB, BE and UB cases in these figures, the LB case is found to provide the largest seismic demands followed by the BE and UB cases. This can be attributed to the combined effect of the larger dynamic earth pressures applied on the deeply embedded structure supported by the softer soil layers. On the other hand, the vertical and in-plane shear forces are generally largest for the UB case, particularly at the transition between the above grade and below grade portion of the wall where

the stiffer UB case subgrade profile provides a stronger support resulting in larger forces in the exterior RB wall.

The plots in Figure 9B.7-15 through Figure 9B.7-17 show that the out-of-plane moments on the RB exterior wall are of a small magnitude with peak amplitudes at the connections of the RB exterior wall with the interior floors and wing walls. An abrupt reduction in force magnitudes is noticed in these figures below the depth of 24.4 m where the higher magnitude dynamic earth pressures from the softer in-situ soils end and the much lower magnitude dynamic pressures from the stiff the rock formation start.

The RB cylindrical shape together with the softer connections between the SCCV and RB walls and slabs help reduce the demands on the RB wing walls although these are located on the load path for resisting earth pressures. This is demonstrated in Figure 9B.7-18 through Figure 9B.7-20 where wing walls are found to be subjected to relatively lower in-plane shear forces than the RB exterior wall which resists most of the dynamic earth pressure loads.

#### **9B.7.5.2 Seismic In-Structure Responses**

5% damped ISRS for responses at key nodal locations obtained from the analysis of the LB, BE and UB subgrade profile cases are compared as shown in Figure 9B.7-21 and Figure 9B.7-22 to evaluate the effect of subgrade variations on the RB in-structure response. As shown in these figures, these ISRS are also compared to the corresponding enveloped and broadened ISRS developed as described in Section 9B.7.1.2 to determine whether the broadened spectra can capture possible shifts of the ISRS peak frequencies.

The comparisons show that the first ISRS peaks are at frequencies that coincide with the SSI system response dominant frequencies listed in Subsection 9B.7.4. In the high-frequency range, at frequencies higher than 8 Hz and 20 Hz in the horizontal and vertical direction, respectively, and for the overall largest ISRS amplitude, the UB case provides the highest ISRS.

One exception is the vertical ISRS for the vertical response at the top of the pedestal, shown in Figure 9B.7-21, where the analysis of the LB subgrade profile has an ISRS peak that exceeds the ISRS obtained from the analysis of the UB and BE subgrade profile cases. This peak response is a result of resonance between the structural frequency of the pedestal and the secondary dynamic mode of the LB subgrade profile at that frequency. Nevertheless, this peak ISRS frequency of 16 Hz is significantly lower than the cutoff frequency of 22 Hz used for the analysis of the LB subgrade profile, thus demonstrating that the selected cutoff frequency for the LB profile analysis is adequate for the calculation of enveloping SSI responses.

#### **9B.7.5.3 Seismic Relative Displacements**

Relative displacements obtained from the analysis of the LB, BE and UB subgrade profile cases are compared to evaluate the effect of subgrade variations on the seismic displacement response of the RB integrated structures.

Table 9B.7-4 presents the estimated maximum relative displacements at the operating floor slab (Node 106595; 4.9 m above grade), at the centre of the roof (Node 129688; 30.5 m above grade) and at top of the RPV pedestal (Node 89097; 8.3 m below grade). As can be seen from this table, the largest relative horizontal and vertical displacements are provided by the BE and UB cases.

Considering that the magnitudes of the estimated relative displacements are very small, smaller than 20 mm at the RB roof, the observed response demonstrates the relatively stiff structural characteristics of the RB.



### **9B.7.6 Sensitivity Analysis Cases**

Following Chapter 3, Subsection 3.3.1.2.4, sensitivity evaluations are performed to investigate the effects of structural stiffness variation, soil-structure interface conditions, soil separation and groundwater variations, and excavation and backfill properties on the seismic response and design of RB SSC.

The sensitivity evaluations are based on results of sensitivity SSI analyses performed using the BE subgrade profile properties and SSI analysis models representing conditions that bound the variation of the evaluated SSI parameter. The sensitivity evaluations are performed by comparing in-structure responses at key locations and stress demands for key structural members within the RB. Subsection 9B.7.6.1 through Subsection 9B.7.6.4 summarize the findings of these sensitivity evaluations.

Per guidance of NEDO-33914, Section 5.3, the results of the sensitivity analysis are included in the RB seismic design basis when comparisons show significant exceedances, greater than 10%, in the RB seismic response due to any of the examined effects. To do so, results of the sensitivity cases are enveloped with those of the design basis analyses of LB, BE and UB subgrade profiles. The enveloped results are then used to develop seismic member force and moment demands, and design ISRS, following the methodology in Subsections 9B.7.1.1 and 9B.7.1.2 to provide a bounding basis for the site-specific seismic design of the Seismic Category A and B SSC at the DNNP site. Subsection 9B.7.6.5 provides a sample of bounding seismic design basis ISRS developed to account for observed exceedances at a given location.

#### **9B.7.6.1 Structural Stiffness Variation Effects on Responses**

As stated in Appendix 9B.6, Subsection 9B.6.1.2, shell elements modeling Steel Bricks™ members of the RB integrated structures are assigned BE effective stiffness properties and Level 1 damping value of 3% in the structural model used for the design basis seismic analysis cases. To evaluate the structural stiffness variation effects on the integrated RB responses, a sensitivity case is performed on an SSI analysis model representing LB stiffness properties of integrated RB structures. In this model, all Steel Bricks™ members are considered fully cracked in flexure and in-plane shear and are assigned a higher level 2 damping value of 5% to reflect the higher level of energy dissipation as a result of concrete cracking. Both analyses are performed for the BE subgrade profile using structural models with effective stiffness properties.

The comparisons of stress responses obtained from the models with BE (Effective) and LB (Reduced) stiffness properties at the RB exterior wall, the RPV pedestal and at the SCCV top slab show that the overall effects of stiffness reduction in Steel Bricks™ members are small. The comparisons show very small differences in the magnitude and distribution of the membrane force and out-of-plane moment demands on the RB exterior wall as demonstrated by Figure 9B.7-23 and Figure 9B.7.24. Figure 9B.7-25 indicates minimal differences in the out-of-plane bending magnitudes at the SCCV top slab due to the structural stiffness reduction. The most noticeable differences due to stiffness reduction, albeit limited, are observed in the stress demands on the RPV pedestal, where a small increase in vertical forces and in-plane shear forces is observed at the bottom half of the pedestal as shown in Figure 9B.7-26 and Figure 9B.7-27.

5% damped ISRS in the three orthogonal directions obtained from the “Effective Stiffness” and “Reduced Stiffness” models are compared at the operating floor slab, at the top of the RPV pedestal and at the edge of the RB roof to assess the amplifications and peak frequency shifts in the in-structure responses induced by concrete cracking. These ISRS are also compared with the broadened 5% damped ISRS, developed as described in Subsection 9B.7.1.2 as the envelope of the three design basis SSI analyses of LB, BE and UB subgrade profiles, to determine if the enveloping and broadening bound the shift in peak response frequencies.

Comparisons indicate small shifts of peak responses to lower frequencies in the horizontal direction, with a few of the exceedances surpassing 10% at the peak responses. These exceedances are nevertheless mostly enveloped by the bounding ISRS. In the vertical direction, similar exceedances are also observed, with the ISRS at the operating floor the most notable at a frequency of about 13 Hz (See Figure 9B.7-28), where a shift to a lower peak frequency occurs reflecting the softer “Reduced Stiffness” properties for the slab and supporting structure.

#### **9B.7.6.2 Soil Separation Effects on Responses**

The linear elastic SSI analyses use models that assume a perfect bond between the embedded RB exterior wall and the subgrade, where rigid springs connect the RB exterior wall to the surrounding soil layers. Considering the possibility of the RB exterior wall separating in parts from the surrounding soil or rock over short instances of time, a sensitivity evaluation is performed following the guidance of NEDO-33914, Section 5.3.9, to investigate the effects of soil separation on the seismic response of the integrated RB at the DNNP site.

To assess the extent of soil separation, the maximum dynamic lateral earth pressures obtained from the design basis SSI analysis of the BE subgrade profile are compared in Figure 9B.7-29 with LB estimates of the lateral static stress in the subgrade and lateral static earth pressure. The depth of approximately 15 m at which the seismic pressure exceeds the static pressure (see Figure 9B.7-29) is adopted as a separation depth for the evaluation of soil separation effects on the seismic response of the RB at the DNNP site.

Both analyses are performed for the BE subgrade profile using structural models with effective stiffness properties. The model for the soil separation sensitivity SSI analysis is developed by reducing the stiffness of the springs connecting the RB exterior wall to the surrounding subgrade from finished grade down to the depth of 15 m to a very small value.

Comparisons of stress responses at the RB exterior wall, the RPV pedestal and SCCV top slab obtained from the “Bonded” and “Separated” models show that, as expected, the tangential (hoop) force on the RB exterior wall below grade decreases due to soil separation. On the other hand, the soil separation can result in significant increases in the vertical membrane (25%) and in-plane shear (70%) forces on the wall portion between grade level and depth of separation (See Figure 9B.7-30 and Figure 9B.7-31). This increase is attributed to the transfer of the dynamic forces from the top 15 m of soil including from the surrounding Power Block structures to the exterior wall portion that is fully bonded with soil starting at a depth of 15 m.

The effects of soil separation on the SCCV top slab and RPV pedestal are minimal as demonstrated by Figure 9B.7-32 and Figure 9B.7-33.

To assess possible amplifications and peak frequency shifts in the in-structure responses due to soil separation, comparisons are made of 5% damped ISRS obtained at the top of the RPV pedestal, at grade elevation of the RB exterior wall and at the edge of the RB roof from the “Bonded” and “Separated” models. The broadened 5% damped ISRS, developed as described in Subsection 9B.7.1.2 as the envelope of the three design basis SSI analyses of LB, BE and UB subgrade profiles, are also compared with the ISRS from the “Separated” model to determine if the enveloping and broadening bound the shifts in peak response frequencies.

The comparisons of horizontal ISRS at all three locations show shifts of the ISRS peaks to slightly lower frequencies that are due to the reduced depth of support provided by the subgrade embedment. The frequency shifts in the vertical ISRS peaks are less pronounced. In general, the shifts in peak frequency responses are bounded by the broadening of the ISRS calculated as envelope of the three design basis SSI analyses.

Noticeable ISRS peak amplifications of more than 10% are observed in the horizontal and vertical responses of the RB exterior wall at grade presented in Figure 9B.7-34.

The effects of soil separation on the horizontal responses inside of the RB structure are not significant. However, the frequency shift in the vertical response of the RPV pedestal shown in Figure 9B.7-35 is significant (>40%) due to resonance effects. Figure 9B.7-35 shows that this amplification in the vertical ISRS is bounded by the enveloping and broadening of the design basis analyses ISRS.

#### **9B.7.6.3 Groundwater Variation Effects on Responses**

The design basis seismic analysis cases of the RB reflect fully saturated conditions for all soil materials located below the DNNP site-specific nominal groundwater table elevation that is located 3 m below the plant grade (See Chapter 3, Subsection 3.3.1.1.6). The saturated soil dynamic properties are simulated by increasing the Poisson's ratios to reflect the compression wave velocity of water in the LB, BE and UB compression wave velocity subgrade profiles used for the design basis analyses.

A sensitivity evaluation is performed following the guidelines of NEDO-33914, Section 5.3.10 to examines the potential effects of groundwater level variability on seismic design. The evaluation is based on results of sensitivity SSI analysis of the BE soil profile reflecting unsaturated conditions for all soil materials down to the surface of the first rock, at a depth of 24 m from the plant grade.

Unsaturated soil conditions are simulated by adjusting the Poisson's ratios of the soil materials resulting in compression wave velocities of soil materials lower than those in the BE subgrade profile used for the design basis analysis as shown in Figure 9B.7-36.

Stress responses at the RB exterior wall, the RPV pedestal and SCCV top slab obtained from the "Saturated" and "Unsaturated" subgrade profiles are compared. The comparisons show that the smaller values of the Poisson's ratios for the soil materials in the unsaturated BE profile reduce the lateral support these soil layers provide to the RB exterior wall, which increases seismic stress demands on the below grade portion of the wall as shown in Figure 9B.7-37 and Figure 9B.7-38. Figure 9B.7-39 shows significant increases in moment demands on the SCCV top slab which are associated with the response in the vertical direction. The effects of groundwater variation on the membrane forces in the RPV pedestal are minimal as demonstrated by Figure 9B.7-40.

To assess possible amplifications and peak frequency shifts in the in-structure responses due to variations of groundwater table, comparisons are made of 5% damped ISRS obtained for the operating floor slab, the top of the RPV pedestal and at grade elevation of the RB exterior wall from the "Saturated" and "Unsaturated" BE subgrade profiles. The broadened 5% damped ISRS, developed as described in Subsection 9B.7.1.2 as the envelope of the three design basis SSI analyses of LB, BE and UB subgrade profiles representing saturated soil conditions, are also compared with the ISRS from the "Unsaturated" subgrade profile to determine if the enveloping and broadening bound the shifts in peak response frequencies.

The comparisons of the horizontal ISRS results show that the effects of the groundwater table variation at the DNNP site on the response of the BWRX-300 RB in the horizontal direction are minimal as shown in Figure 9B.7-41 and Figure 9B.7-42. This is because the shear-wave velocities that govern the horizontal SSI response are not affected by the groundwater table. Limited exceedances are observed in Figure 9B.7-41 in the horizontal response of the RB exterior wall at grade at frequencies greater than 10 Hz.

The comparisons of the vertical ISRS in Figure 9B.7-41 and Figure 9B.7-42 show that the effect of groundwater table variation on the response to the vertical motion can be significant. This is because the vertical SSI responses are affected by the compression wave velocities of the softer soil materials that are affected by the unsaturated conditions. The figures indicate that the consideration of unsaturated soil conditions through the depth of the embedment can result in significant (>10%)

increases in peak ISRS amplitudes, in particular for the ISRS representing out-of-plane slab responses, such as seen in Figure 9B.7-42 for the operating floor slab.

#### **9B.7.6.4 Excavation Support and Backfill Effects on Responses**

As stated in Chapter 3, Subsection 3.5.1.1.2, the BWRX-300 design conservatively does not consider the resistance provided by the excavation support or the lean concrete fill used to backfill the gaps between the RB exterior wall and the excavated soil and rock. The excavation support is designed for temporary use and is excluded from the SSI analysis models because the excavation support and the concrete fill are not expected to maintain their structural integrity throughout the operational life of the plant. A sensitivity evaluation is performed following the guidelines of NEDO-33914, Section 5.2.8, to examine the potential effects the presence of the excavation support and lean concrete fill may have on the seismic response and design of the deeply embedded RB at the DNNP site.

As noted in Appendix 9B.6, Subsection 9B.6.2.1, a ring of solid elements is included in the SSI analysis model around the RB exterior wall below grade to represent the backfill material. For the design basis SSI analysis cases, the near-field elements are assigned the same properties as the surrounding far-field subgrade model. To examine the effects of excavation support and concrete fill on the seismic response of the RB, a sensitivity analysis is performed on an SSI model where the near-field solid elements are assigned lean concrete properties. Since the design of the excavation support has not been finalized yet, this is considered an approximation for simulating stiffer backfill and excavation support conditions. Both analyses are performed for BE subgrade profile using structural models with effective stiffness properties.

Stress responses at the RB exterior wall, the RPV pedestal and SCCV top slab obtained from the design basis SSI model without excavation support and backfill and the sensitivity SSI analysis model with excavation support and backfill are compared. The comparisons show that the additional lateral stiffness provided by the near-field elements simulating concrete fill and excavation support results in a significant decrease in the seismic stress demands on the below grade portion of the RB exterior wall (See Figure 9B.7-43 and Figure 9B.7-44). This decrease is attributed to the stronger lateral support provided to the RB by the temporary construction elements. The decrease in stress demands extends to the containment internal structures, such as the RPV pedestal, as shown in Figure 9B.7-45. The effects of the excavation support and backfill on the stress demands in the SCCV top slab out-of-plane responses that are governed by the RB response in the vertical direction are minimal as shown in Figure 9B.7-46.

To assess possible amplifications and peak frequency shifts in the in-structure responses due to the presence of the temporary construction elements, comparisons are made of 5% damped ISRS obtained from the analysis of the design basis model without excavation support and backfill and the sensitivity analysis model with excavation support and backfill around the RB exterior wall. The broadened 5% damped ISRS, developed as described in Subsection 9B.7.1.2 as the envelope of the three design basis SSI analyses of LB, BE and UB subgrade profiles, are also compared with the ISRS from the model with excavation support and backfill to determine if the enveloping and broadening bound the shifts in peak response frequencies.

The comparisons of the horizontal ISRS indicate that the stiffness provided by the temporary construction elements generally reduces the in-structure response of the RB resulting in ISRS with smaller amplitudes. The stiffer near-field conditions result in shifts of the horizontal ISRS peaks to higher frequencies that are well bounded by the broadening of the ISRS obtained from the design basis analyses as shown in Figure 9B.7-47.

Significant decreases in the vertical ISRS amplitudes due to the stiffness of the temporary construction elements are also observed in the lower (<10 Hz) frequency range. The comparisons also indicate that the additional stiffness of the near-field elements may amplify the in-structure responses in the

higher frequency range. Figure 9B.7-47 shows a significant amplification of the operating floor slab vertical response with an exceedance of the peak ISRS amplitude at 13 Hz of more than 30%.

#### **9B.7.6.5 Effects of Enhanced Input Ground Motions**

To address the regulatory guidance of DC/COL ISG-017 (Reference 9B.4-50) discussed in Chapter 3, Subsection 3.3.1.2.4 in the SSI analyses of the deeply embedded RB at the DNNP site, the Nuclear Energy Institute check approach specified in NEDO-33914, Section 5.3.4.1 is implemented by applying at the bottom of the RB foundation augmented input control motions compatible to the Rock Design Ground Response Spectrum presented in Chapter 3, Figure 3.3-14.

To evaluate the level of conservatism introduced in the results of the bounding DNNP site-specific SSI analyses using the augmented input ground motions in Chapter 3, Figure 3.3-14, a sensitivity analysis is performed for the BE subgrade profile where the Performance Based Surface Response Spectra (PBSRS) compatible ground motion is applied at the surface of the SSI model. The level of conservatism is evaluated by comparing ISRS results obtained from the analysis of the design basis model with augmented input motion applied at the bottom of the RB foundation with those from the sensitivity analysis with PBSRS compatible motion applied at the ground surface. Comparisons of 5% damped ISRS are performed at the operating floor slab, at the RB exterior wall at grade and at the edge of the RB roof. The broadened 5% damped ISRS, developed as the envelope of the three design basis SSI analyses of LB, BE and UB subgrade profiles as discussed in Subsection 9B.7.1.2, are also compared with the ISRS from the “surface” sensitivity analysis to illustrate the degree of conservatism in the enveloped and broadened ISRS.

As demonstrated in Figure 9B.7-48, the conservatism introduced in the design basis analyses by the use of augmented input motion at the bottom of the RB foundation, results in ISRS with significantly larger amplitudes. Shifts of vertical ISRS peaks to lower frequencies can also be observed in the “surface” analysis results that are well bounded by the broadening of the ISRS obtained from the design basis analyses.

#### **9B.7.6.6 Development of Bounding Seismic Design Basis ISRS for Use in Design of Seismic Category A and B SSC**

As discussed in Subsections 9B.7.6.1 through 9B.7.6.4, some exceedances in terms of stresses and ISRS demands are observed when comparing to the seismic design basis BE case. To account for these exceedances, bounding seismic design basis demands are increased for each of the key locations where exceedances are observed following the methodology discussed in Subsection 9B.7.6.

Figure 9B.7-49 provides an example for the development of a bounding seismic design basis ISRS at the RB exterior wall at grade. At this location, soil separation effects result in an increase of the horizontal design basis ISRS at a frequency of around 3 Hz, whereas soil separation and groundwater variation effects contribute to an increase in the vertical design basis ISRS.

#### **9B.7.7 Bounding DBE Structural Stress Demands**

This section presents the bounding seismic design basis structural force and moment demands for key structural members within the integrated RB structures. These bounding seismic design basis stresses account for effects of variation of important SSI parameters at the DNNP site discussed in Subsection 9B.7.6

##### **9B.7.7.1 Containment Stress Demands**

Figure 9B.7-50 presents the bounding seismic moment demands for the SCCV top slab, while Figure 9B.7-51 and Figure 9B.7-51 present the bounding seismic membrane forces and moment demands for the SCCV wall.

#### **9B.7.7.2 Containment Internal Structures Stress Demands**

Figure 9B.7-53 presents the bounding seismic membrane forces for the RPV pedestal.

#### **9B.7.7.3 Reactor Building Stress Demands**

Figure 9B.7-54 and Figure 9B.7-55 present the bounding seismic membrane forces and moment demands for the RB Exterior Wall, while Figure 9B.7-56 presents the bounding seismic membrane forces for RB wing walls.

#### **9B.7.8 Bounding DBE In-Structure Demands**

This section presents the bounding ISRS developed to serve as input for the seismic design and qualification of Seismic Category A and B SSC at key locations within the integrated RB. These bounding ISRS are developed as described in Subsection 9B.7.1.2 for eight damping levels (0.5%, 2%, 3%, 4%, 5%, 7%, 10% and 15%) and account for effects of variation of important SSI parameters at the DNNP site discussed in Subsection 9B.7.6.

##### **9B.7.8.1 Containment In-Structure Demands**

Figure 9B.7-57 presents the bounding ISRS at the centre of the common mat foundation, at Node 47963 located within the containment boundary.

##### **9B.7.8.2 Containment Internal Structures In-Structure Demands**

Figure 9B.7-58 presents the bounding ISRS at the top of the RPV pedestal (Node 89097), where the RPV is supported.

##### **9B.7.8.3 Reactor Building In-Structure Demands**

Figure 9B.7-59 presents the bounding ISRS at the edge of common mat foundation (Node 50190). Figure 9B.7-60 presents the bounding ISRS for the operating floor slab (Node 106595). Figure 9B.7-61 presents the bounding ISRS at the centre of the RB roof (Node 129688).

#### **9B.7.9 Maximum DBE Displacements**

Table 9B.7-5 presents the amplitudes of the bounding seismic displacements at key locations within the RB integrated structures. These displacements are computed relative to the free-field ground motion defined at the bottom of the RB foundation using the methodology discussed in Subsection 9B.7.1.3.

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B.7-1: Key Nodal Locations**

Node Number	Location Description	Elevation [m]
47693	Centre of common mat foundation	-34.15
50190	Edge of common mat foundation	-34.15
89097	RPV support at top of pedestal	-8.27
101747	RB exterior wall at grade	0.00
106595	Operating floor slab	4.90
125319	Edge of roof	27.38
129688	Centre of roof	30.48

**Table 9B.7-2: Key Structural Members**

Structural member	Description	Critical stress components
SCCV Top slab	Circular slab	Out-of-plane moment
SCCV wall	Cylindrical wall	Membrane forces, in-plane shear, out-of-plane moment
RPV pedestal	Cylindrical wall	Membrane forces, in-plane shear
RB exterior wall	Cylindrical wall	Membrane forces, in-plane shear, out-of-plane moment
Wing walls	Rectangular walls	Membrane forces, in-plane shear

**Table 9B.7-3: Dominant Frequencies for SSI System Response**

Subgrade Profile Case	Horizontal (X) Hz	Horizontal (Y) Hz	Vertical (Z) Hz
LB	1.75	1.73	4.4
BE	3.5	3.6	6.0
UB	5.2	5.2	8.7

**Table 9B.7-4: Effect of Subgrade Variations on Seismic Displacement Responses**

Location	Subgrade Profile Case	Displacement in X Direction (North-South) (mm)	Displacement in Y Direction (East-West) (mm)	Displacement in Z Direction (Vertical) (mm)
Operating Floor Slab	LB	7	7.3	2.5
	BE	<b>9.7</b>	<b>8.2</b>	2.7
	UB	6.7	6.6	<b>3.6</b>
Centre of RB Roof	LB	10.5	11.4	6.6
	BE	16.0	12.8	<b>7.8</b>
	UB	<b>16.9</b>	<b>15.4</b>	7.4
Top of RPV Pedestal	LB	5.4	5.6	1.0
	BE	<b>12.5</b>	8.9	<b>2.0</b>
	UB	10.4	<b>9.6</b>	1.4

**Table 9B.7-5: Maximum Displacement at Key Nodal Locations Relative to Free-field**

Node Number	Location	Elevation (m)	Displacement in X Direction (North-South) (mm)	Displacement in Y Direction (East-West) (mm)	Displacement in Z Direction (Vertical) (mm)
47693	Centre of common mat Foundation	-34.15	0.7	0.7	0.4
50190	Edge of common mat Foundation	-34.15	0.7	0.7	0.4
89097	Top of pedestal	-8.27	12.5	9.6	2.0
101747	Exterior wall	0.0	7.5	6.5	2.7
106595	Operating floor slab	4.90	9.7	8.2	3.6
125319	Edge of RB roof	27.38	16.2	12.7	4.7
129688	Centre of RB roof	30.48	16.9	15.4	7.8



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

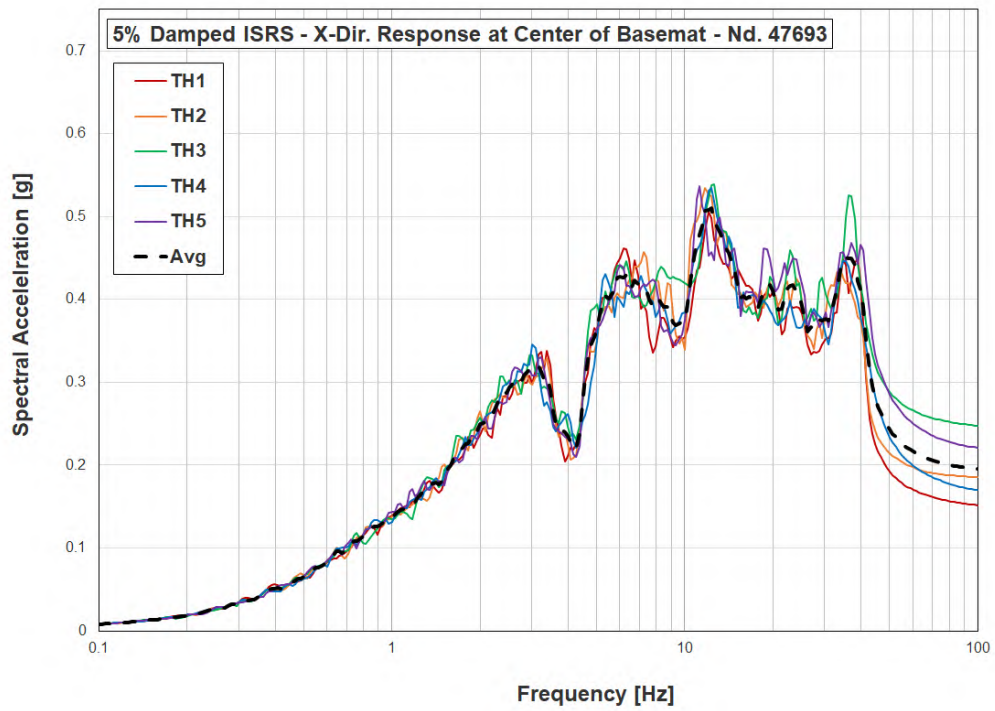


Figure 9B.7-1: Averaging of In-Structure Response Spectra from Five Sets of Acceleration Time History Results

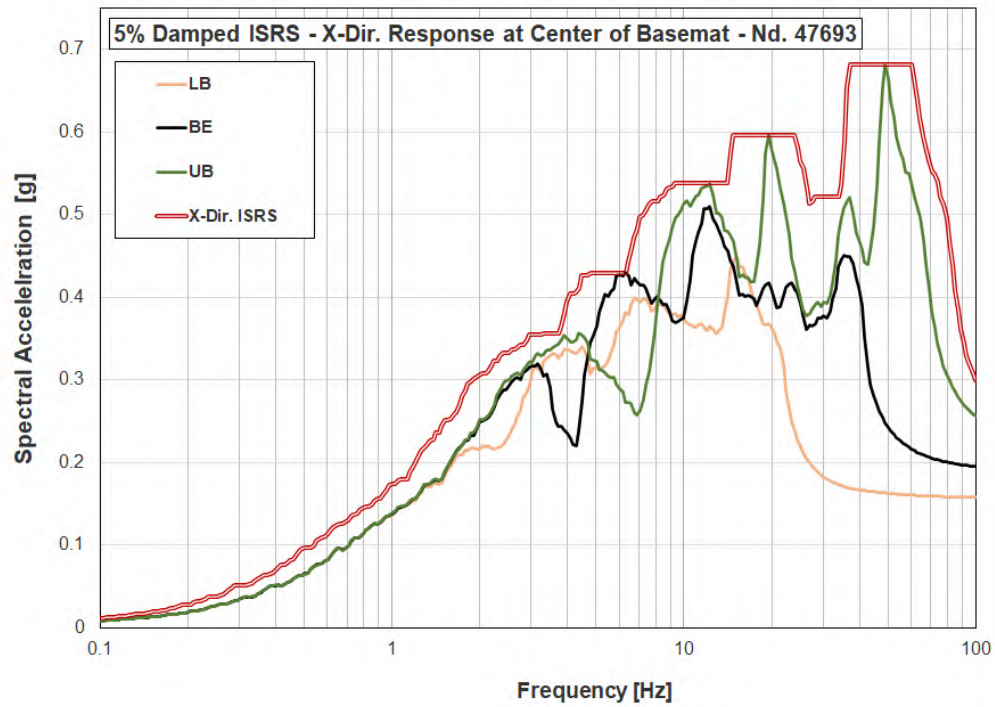
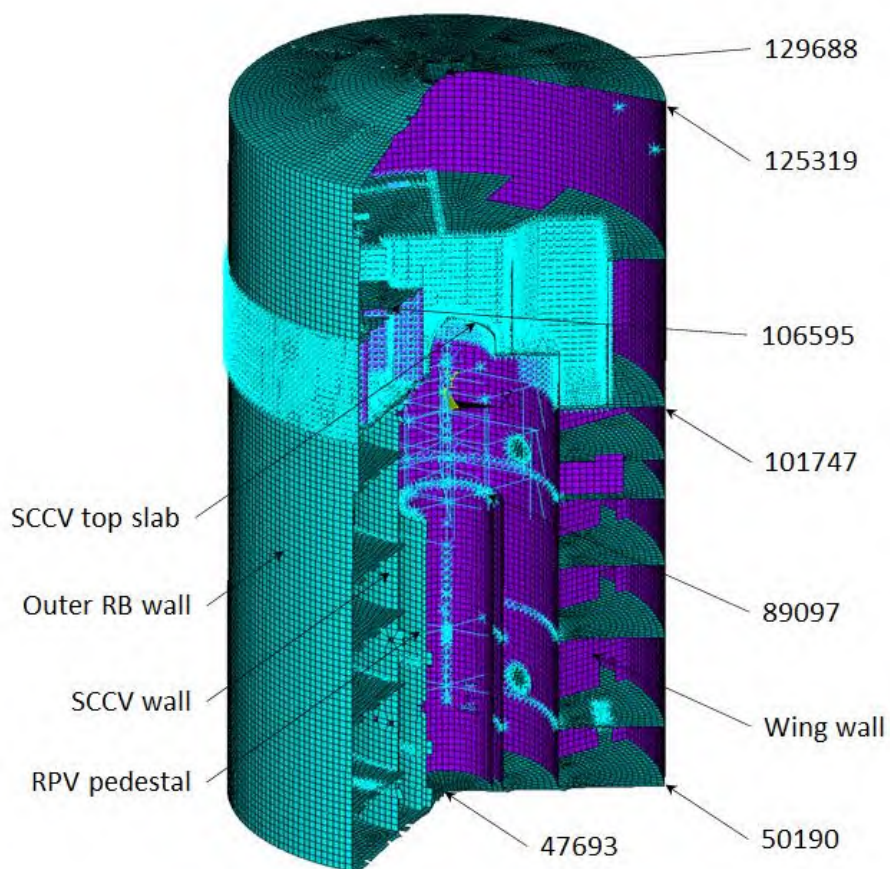


Figure 9B.7-2: Enveloping and Broadening of In-Structure Response Spectra



**Figure 9B.7-3: Key Nodal Locations and Structural Members**

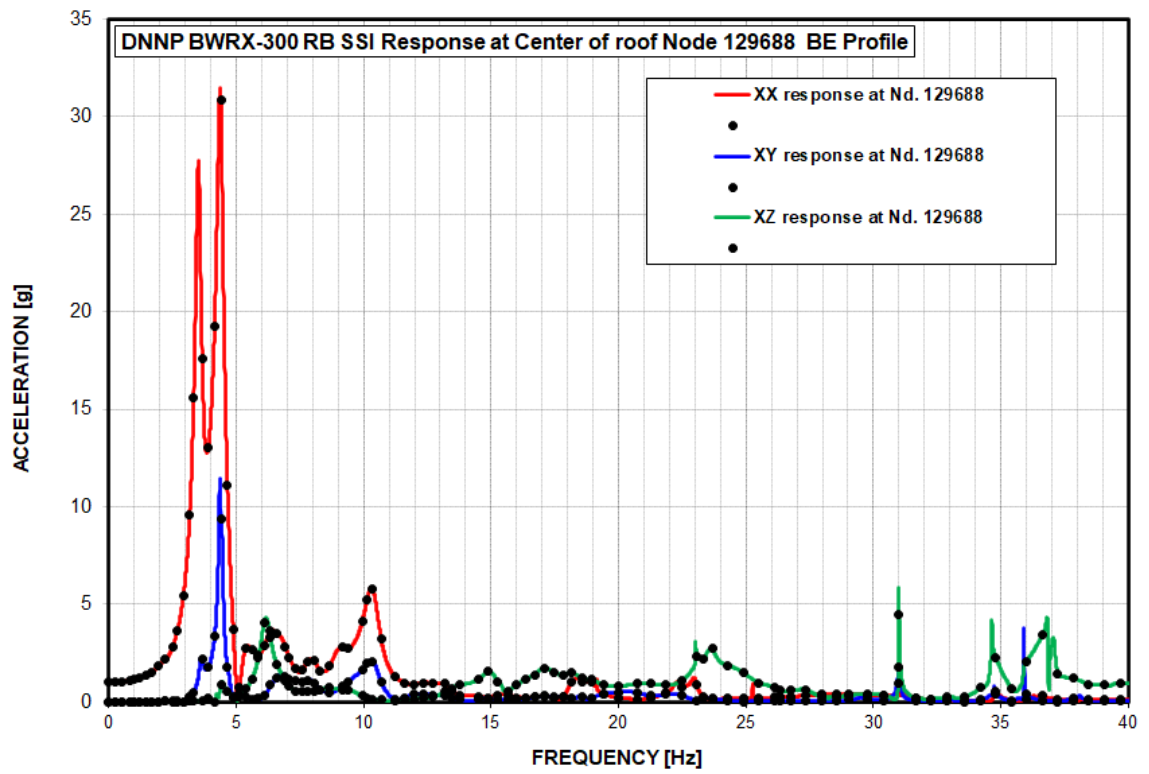


Figure 9B.7-4: Interpolation of Acceleration Transfer Functions

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

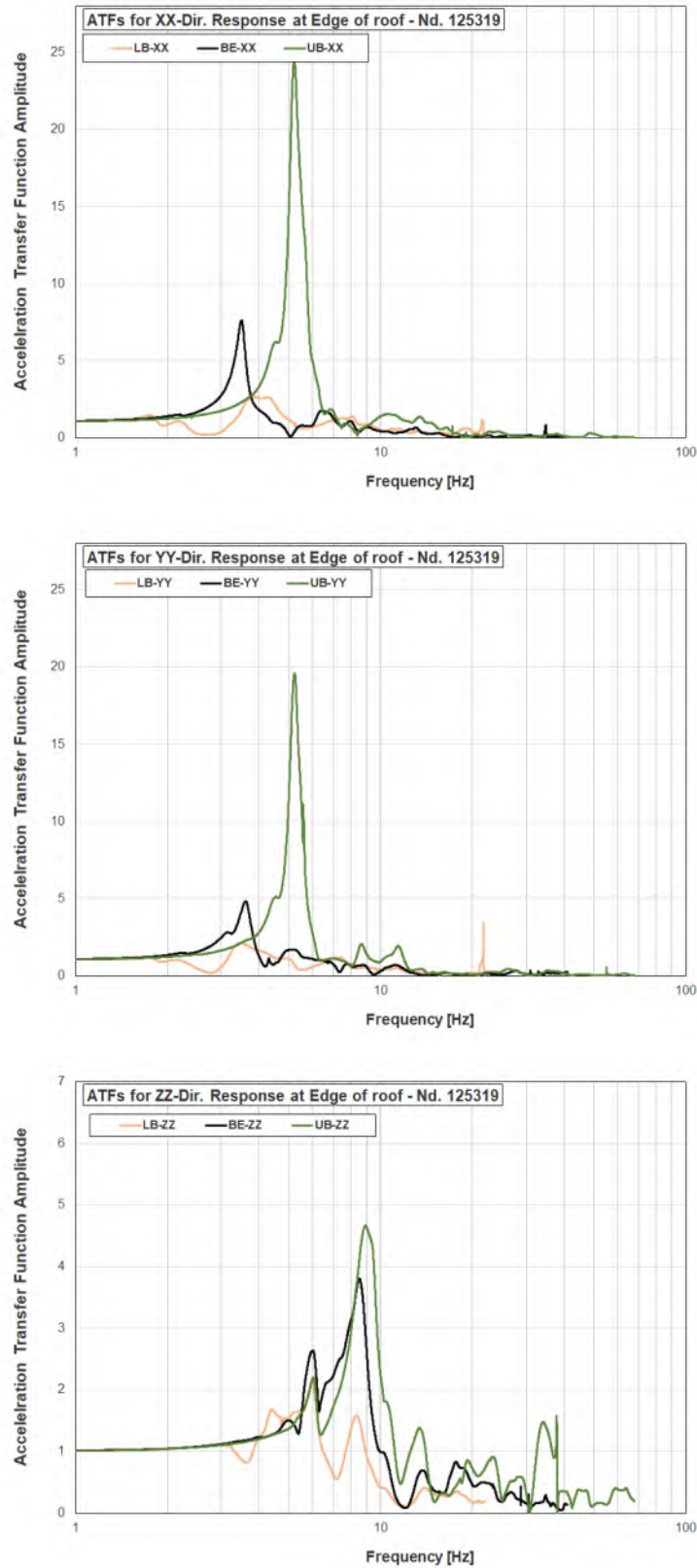
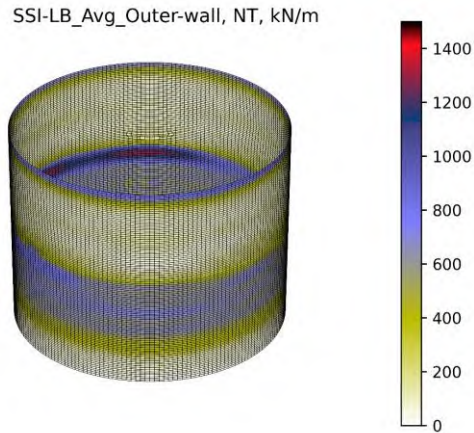
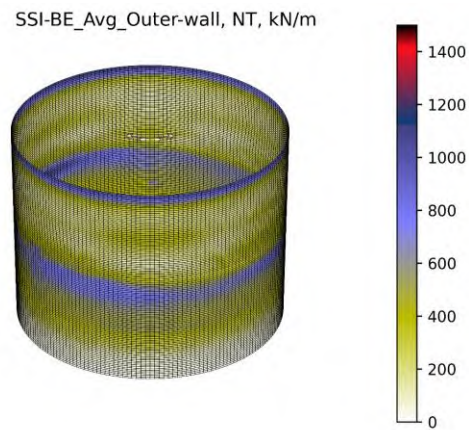


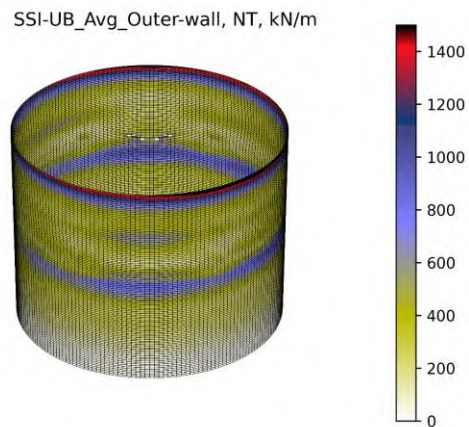
Figure 9B.7-5: Effects of Subgrade Variation on the Normalized Acceleration Time Histories



**Figure 9B.7-6: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – LB Case**



**Figure 9B.7-7: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – BE Case**

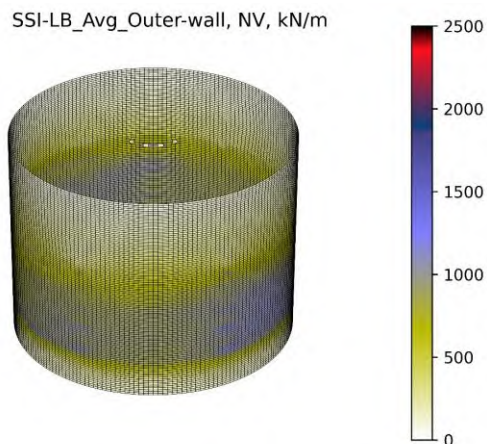


**Figure 9B.7-8: Membrane Tangential (Hoop) Forces – Reactor Building Exterior Wall – UB Case**

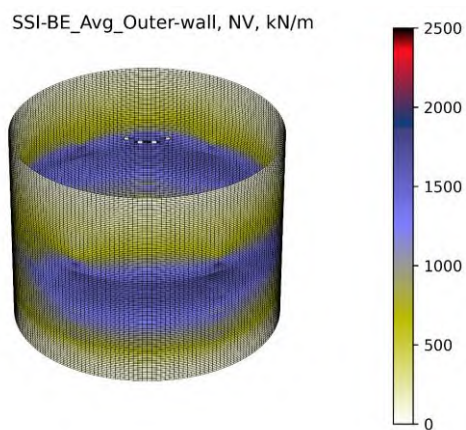
\*Aspect ratio not to scale in these figures



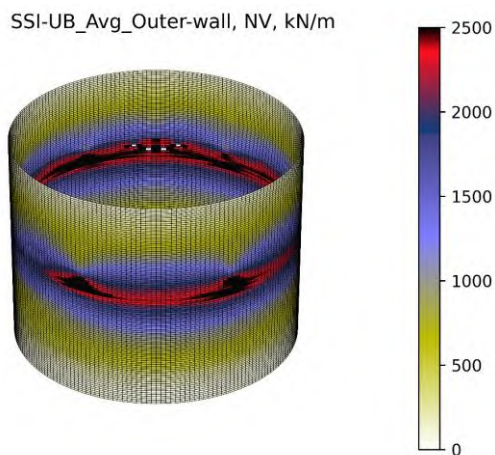
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-9: Membrane Vertical Forces – Reactor Building Exterior Wall – LB Case**



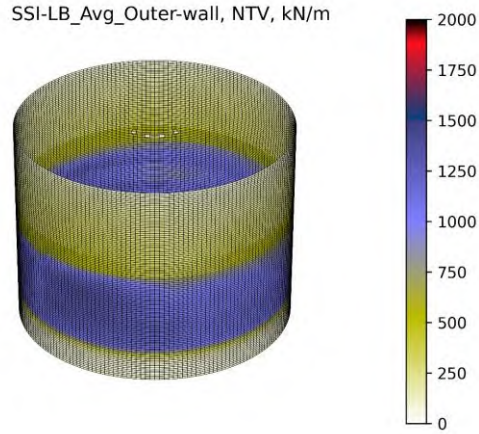
**Figure 9B.7-10: Membrane Vertical Forces – Reactor Building Exterior Wall – BE Case**



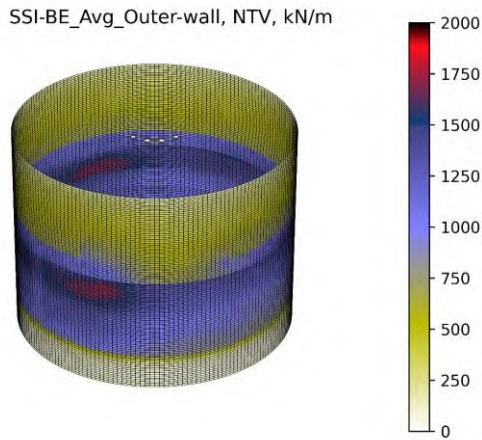
**Figure 9B.7-11: Membrane Vertical Forces – Reactor Building Exterior Wall – UB Case**

\*Aspect ratio not to scale in these figures

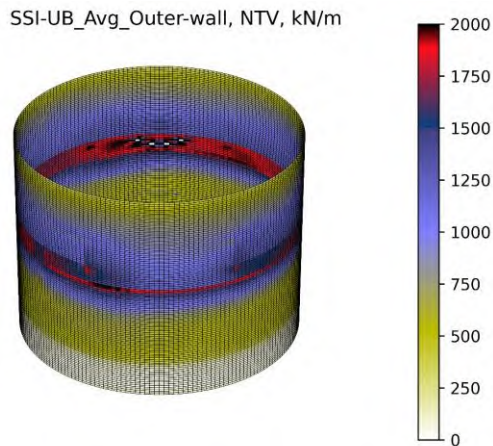
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-12: In-Plane Shear Forces – Reactor Building Exterior Wall – LB Case**

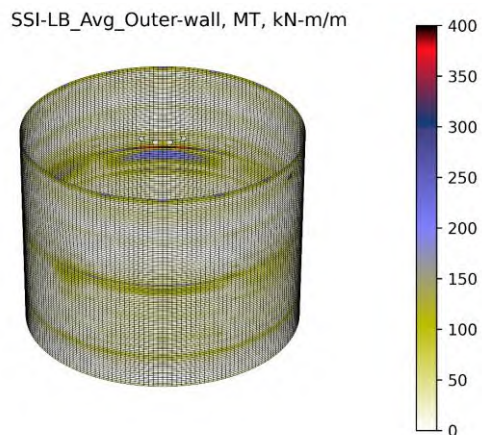


**Figure 9B.7-13: In-Plane Shear Forces – Reactor Building Exterior Wall – BE Case**

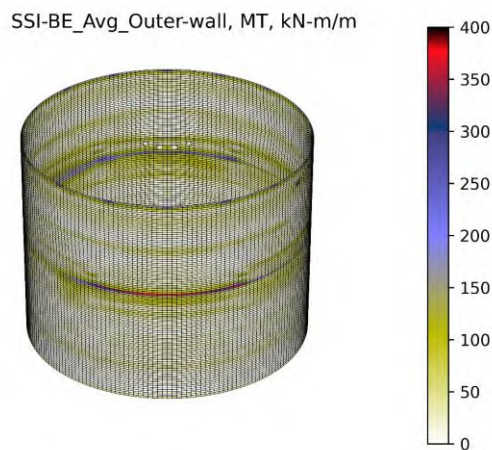


**Figure 9B.7-14: In-Plane Shear Forces – Reactor Building Exterior Wall – UB Case**

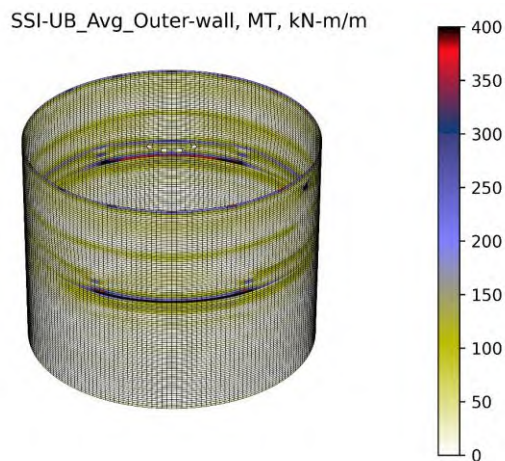
\*Aspect ratio not to scale in these figures



**Figure 9B.7-15: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – LB Case**



**Figure 9B.7-16: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – BE Case**

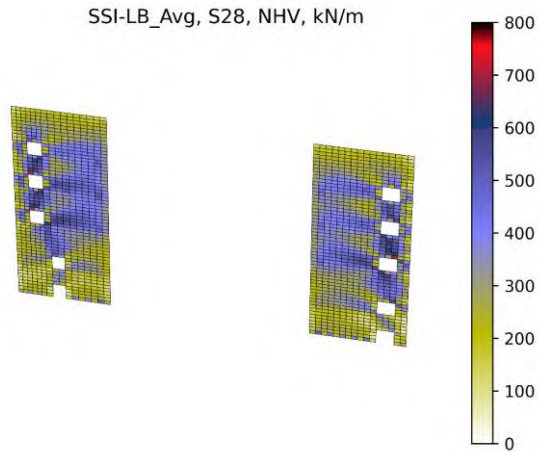


**Figure 9B.7-17: Out-of-Plane Horizontal Moments – Reactor Building Exterior Wall – UB Case**

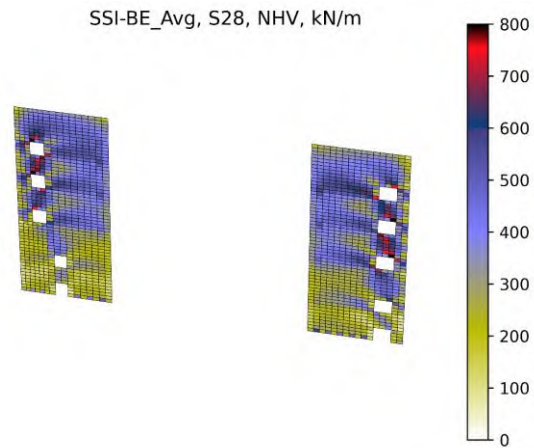
\*Aspect ratio not to scale in these figures



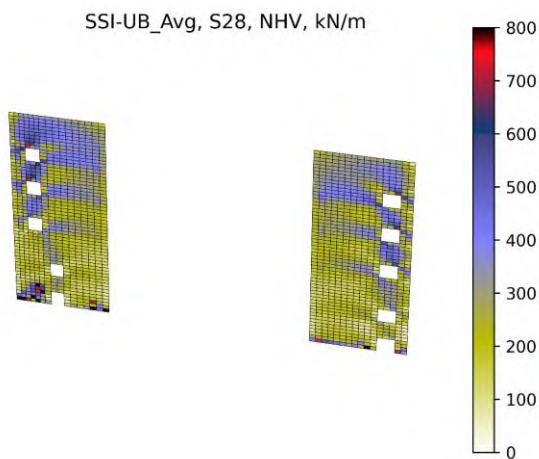
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-18: In-Plane Shear Forces – Wing Walls – LB Case**



**Figure 9B.7-19: In-Plane Shear Forces – Wing Walls – BE Case**



**Figure 9B.7-20: In-Plane Shear Forces – Wing Walls – UB Case**

\*Aspect ratio Not to Scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

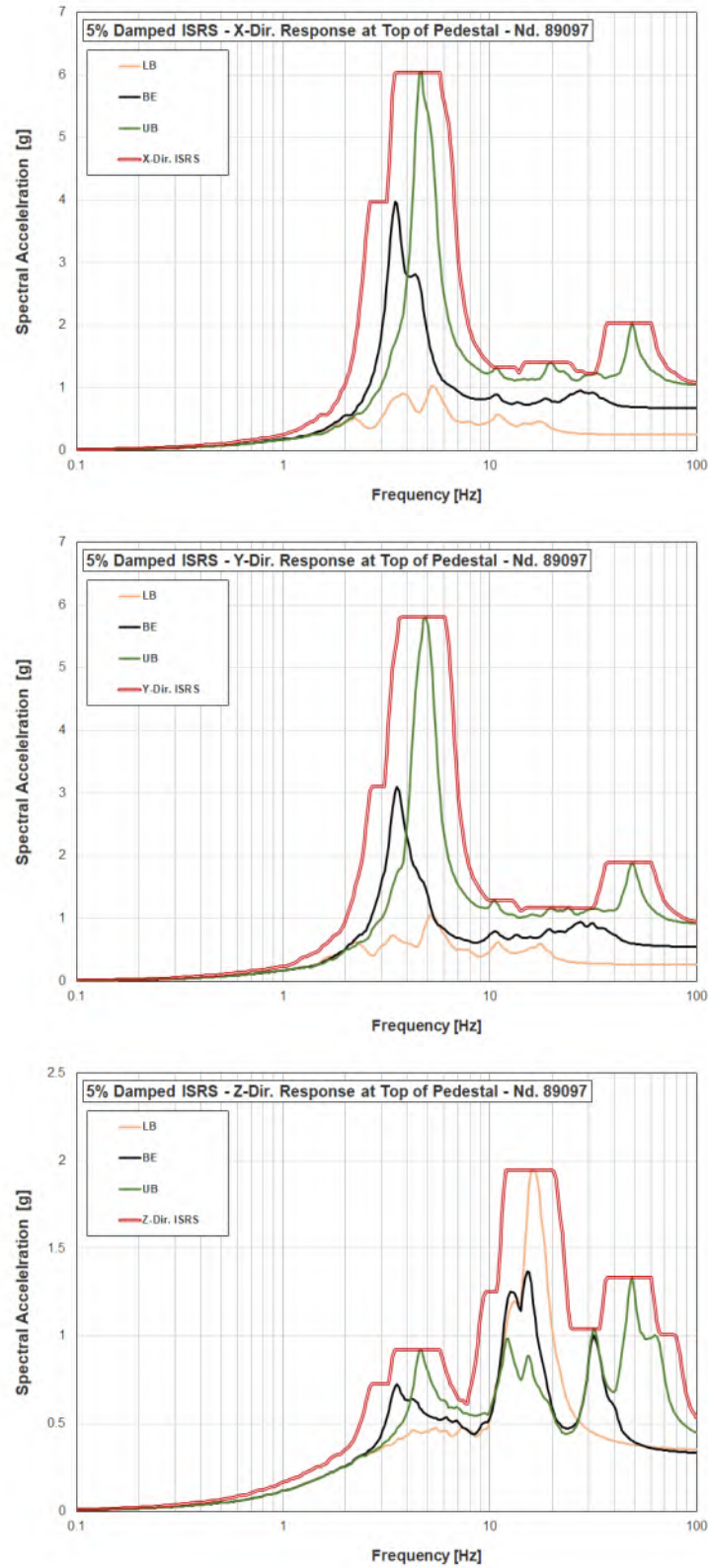


Figure 9B.7-21: Effect of Subgrade Variations on In-Structure Response at Top of Pedestal

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

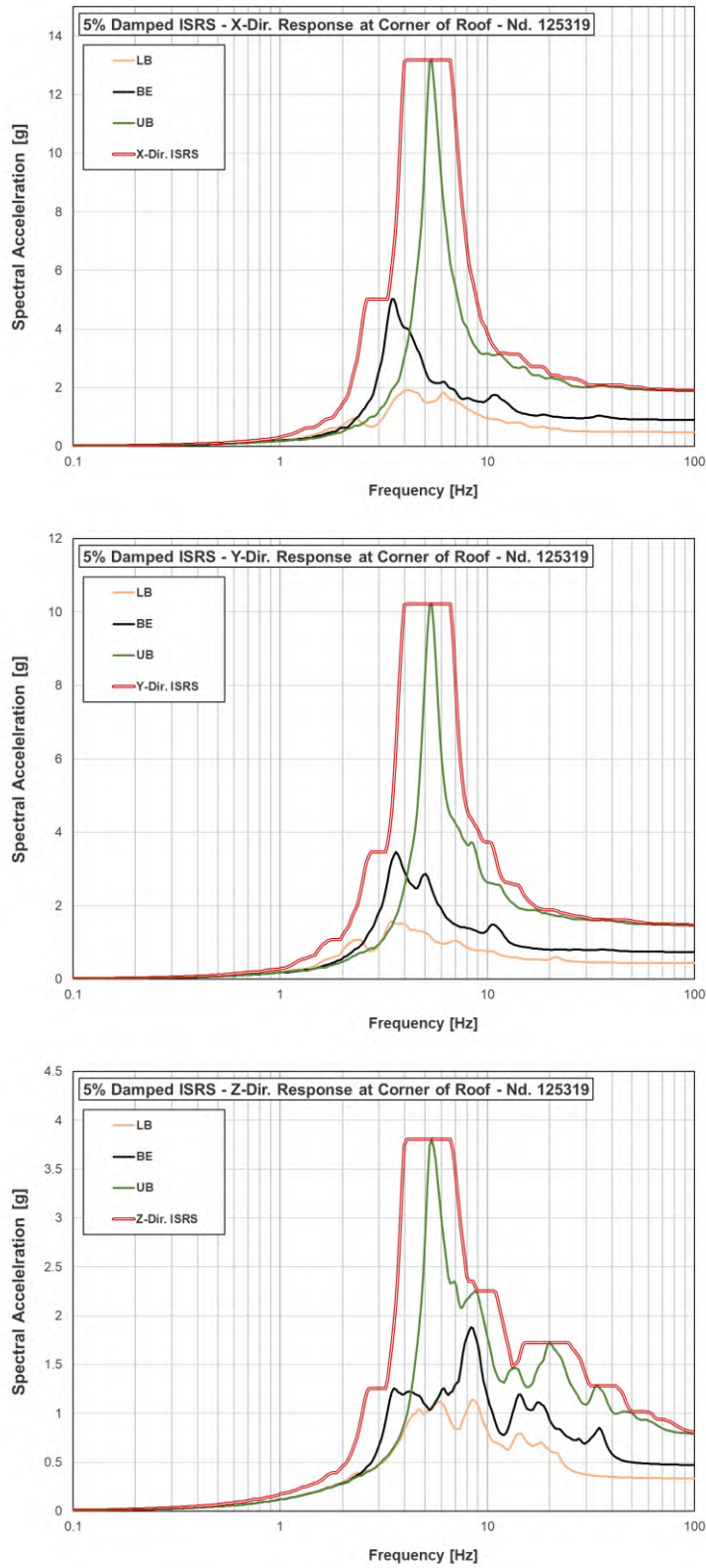
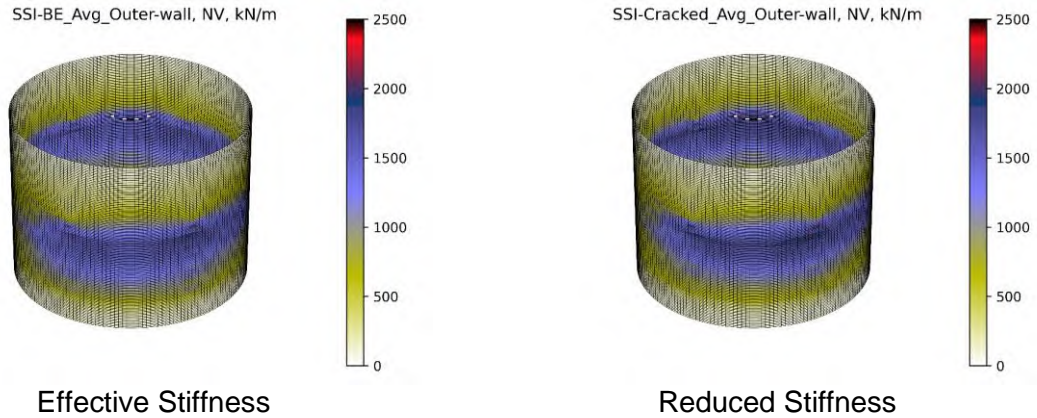
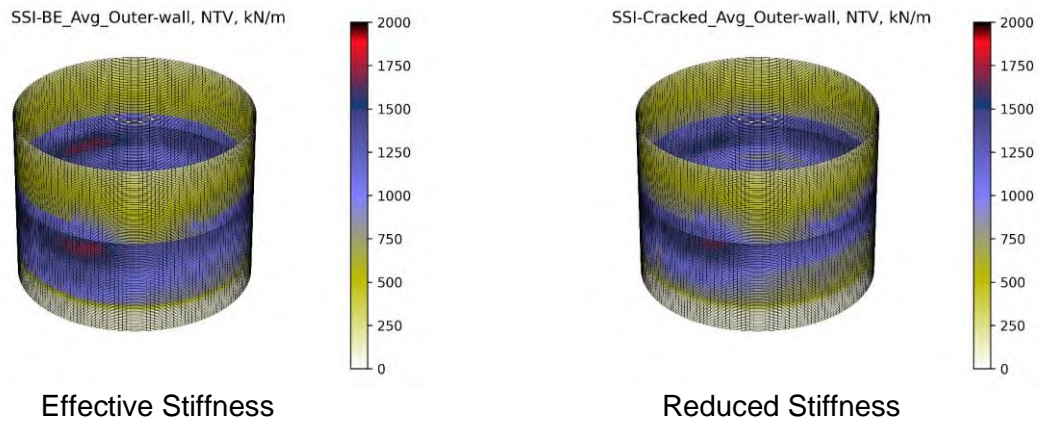


Figure 9B.7-22: Effect of Subgrade Variations on In-Structure Response at Edge of Reactor Building Roof

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



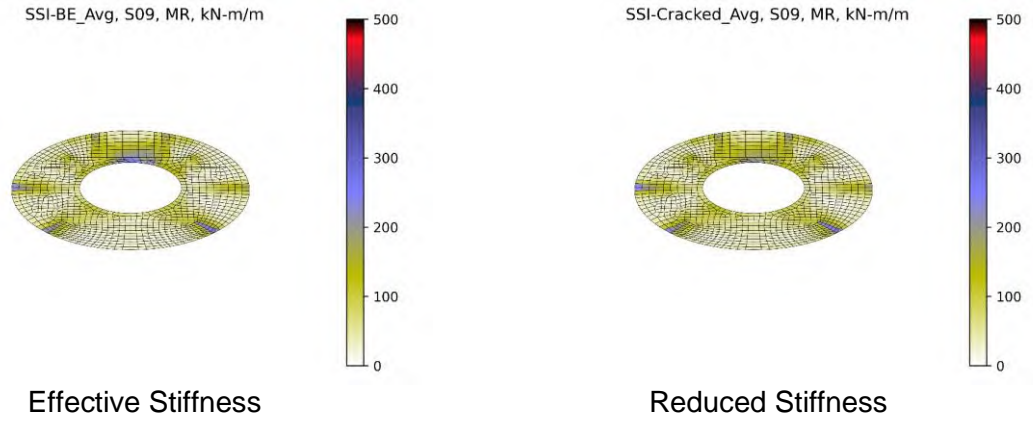
**Figure 9B.7-23: Structural Stiffness Variation Effects on Vertical Forces in the Reactor Building Exterior Wall**



**Figure 9B.7-24: Structural Stiffness Variation Effects on the In-Plane Shear in the Reactor Building Exterior Wall**

\*Aspect ratio not to scale in these figures

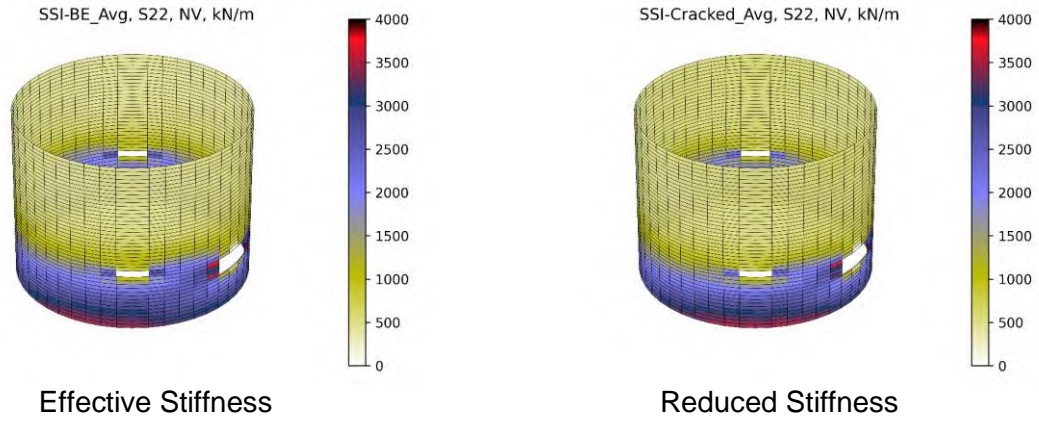
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



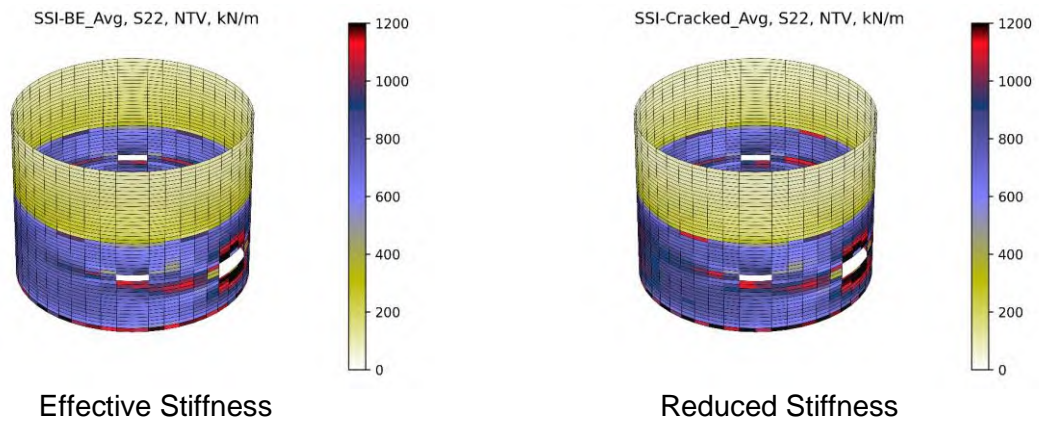
**Figure 9B.7-25: Structural Stiffness Variation Effects on SCCV Top Slab Bending Moment**



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-26: Structural Stiffness Variation Effects on the RPV Pedestal Vertical Force**



**Figure 9B.7-27: Structural Stiffness Variation Effects on the RPV Pedestal In-Plane Shear**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

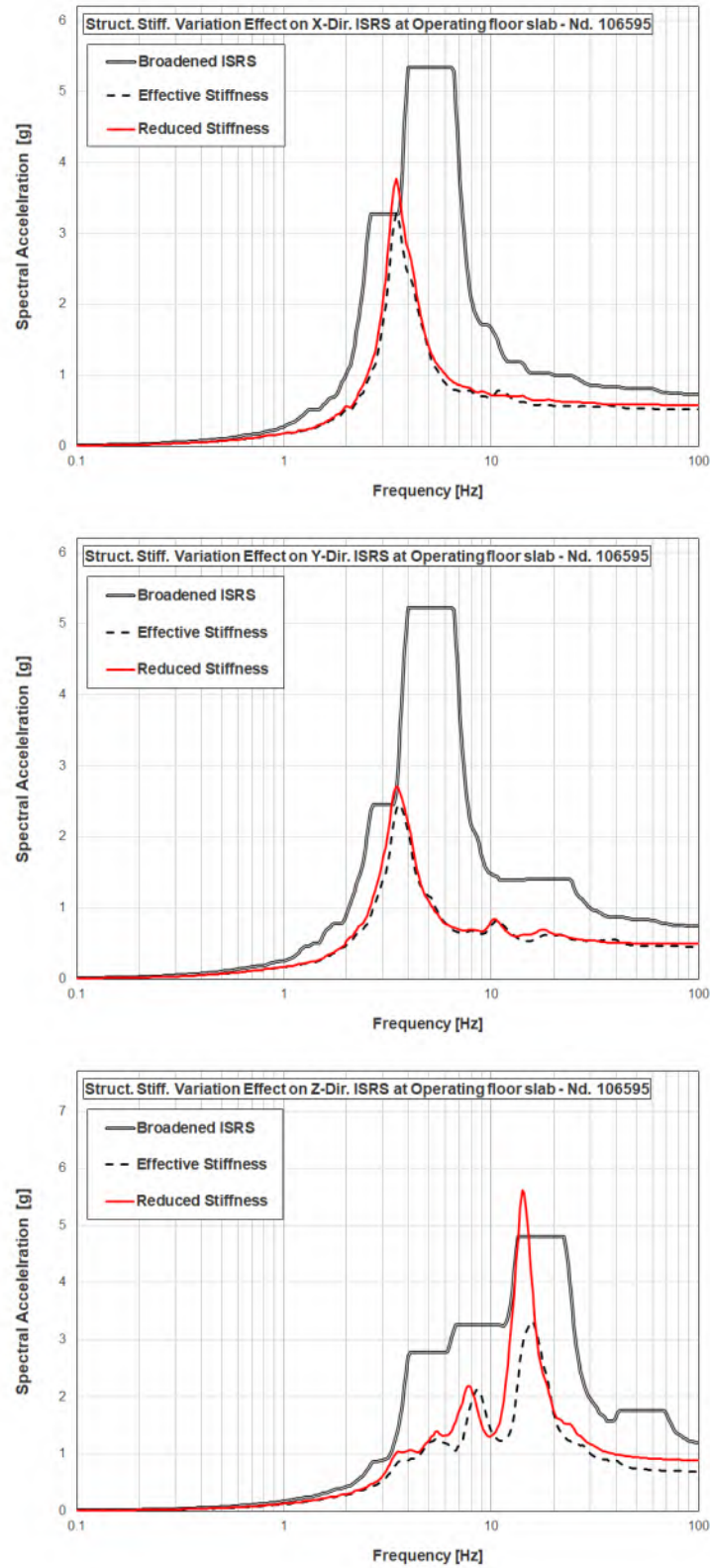


Figure 9B.7-28: Structural Stiffness Variation Effects on In-Structure Response Spectra for Operating Floor Slab

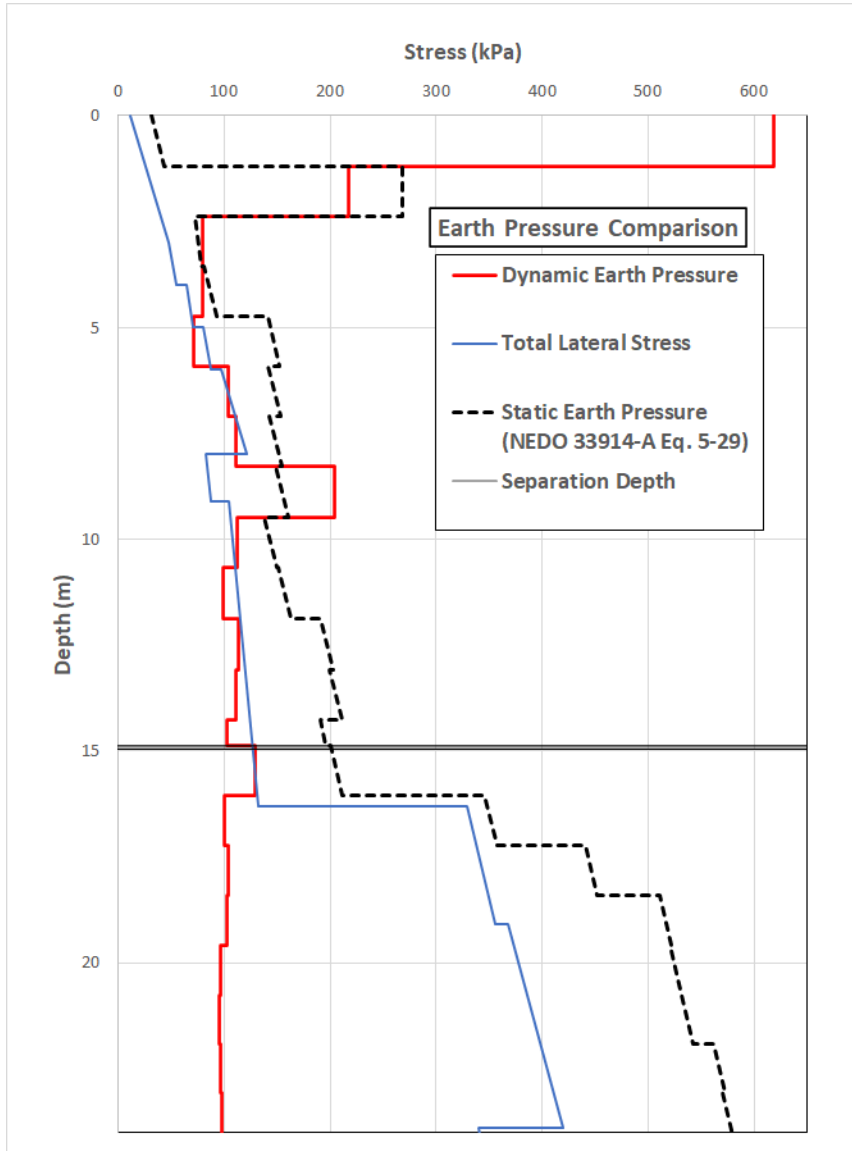
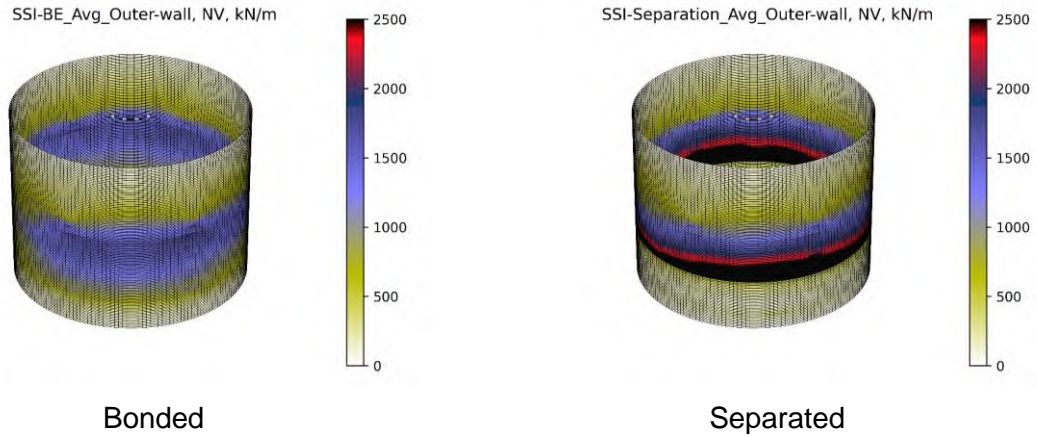


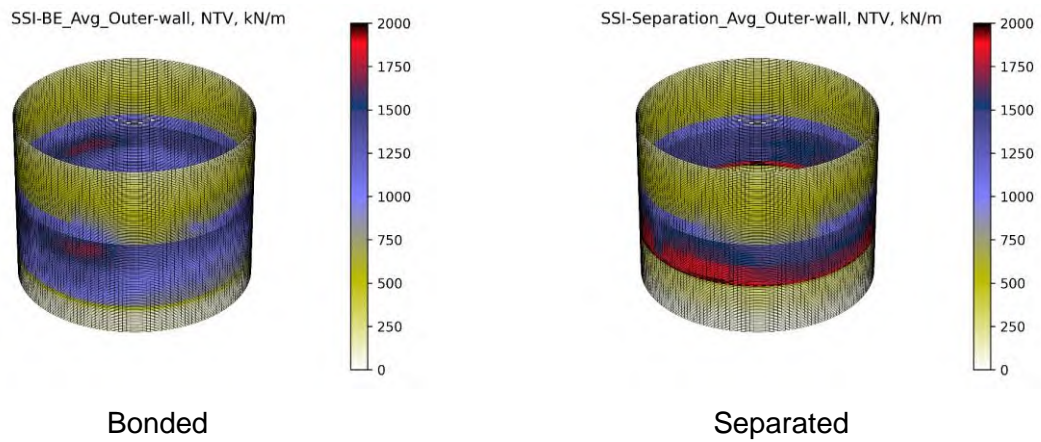
Figure 9B.7-29: Comparison of Dynamic and Static Lateral Earth Pressures



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



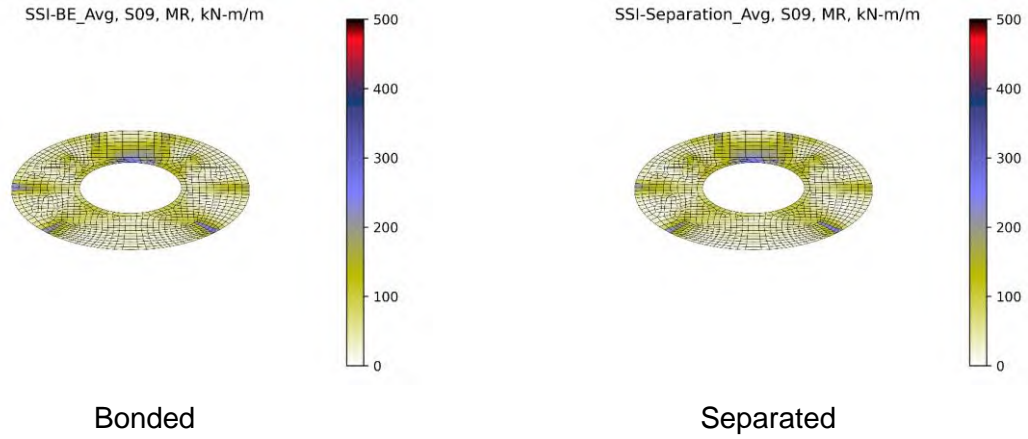
**Figure 9B.7-30: Soil Separation Effects on Vertical Forces in the Reactor Building Exterior Wall**



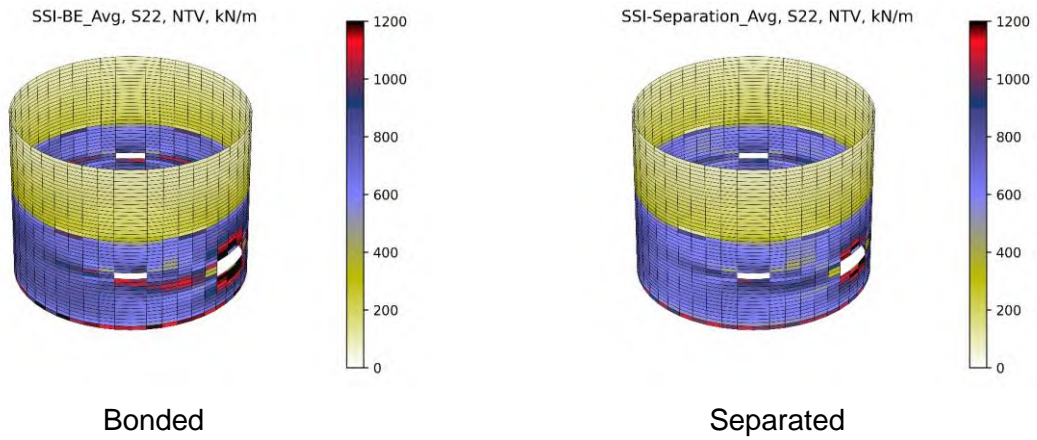
**Figure 9B.7-31: Soil Separation Effects on the In-Plane Shear Force in the Reactor Building Exterior Wall**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-32: Soil Separation Effects on SCCV Top Slab Bending Moment**



**Figure 9B.7-33: Soil Separation Effects on RPV Pedestal In-Plane Shear**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

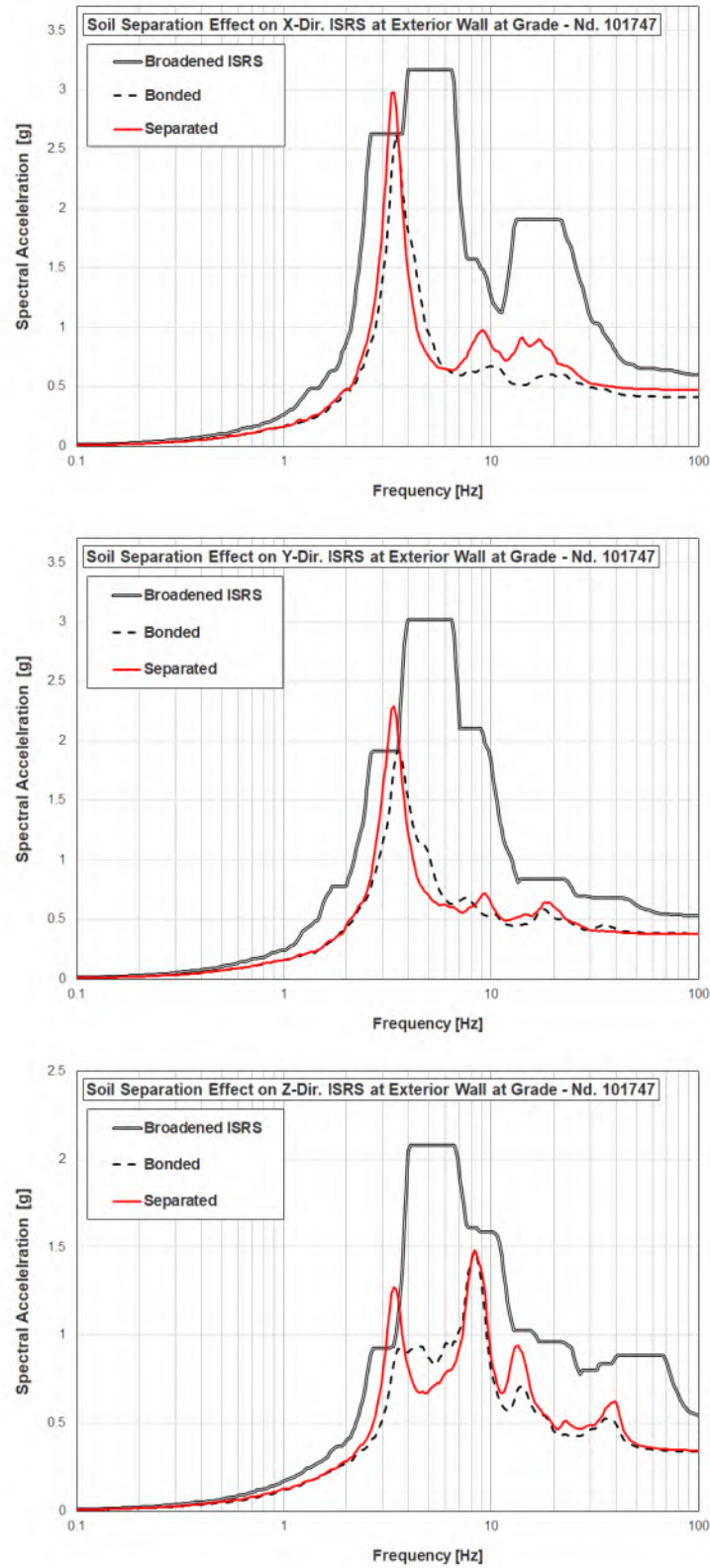


Figure 9B.7-34: Soil Separation Effects on In-Structure Response Spectra for the Reactor Building Exterior Wall at Grade

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

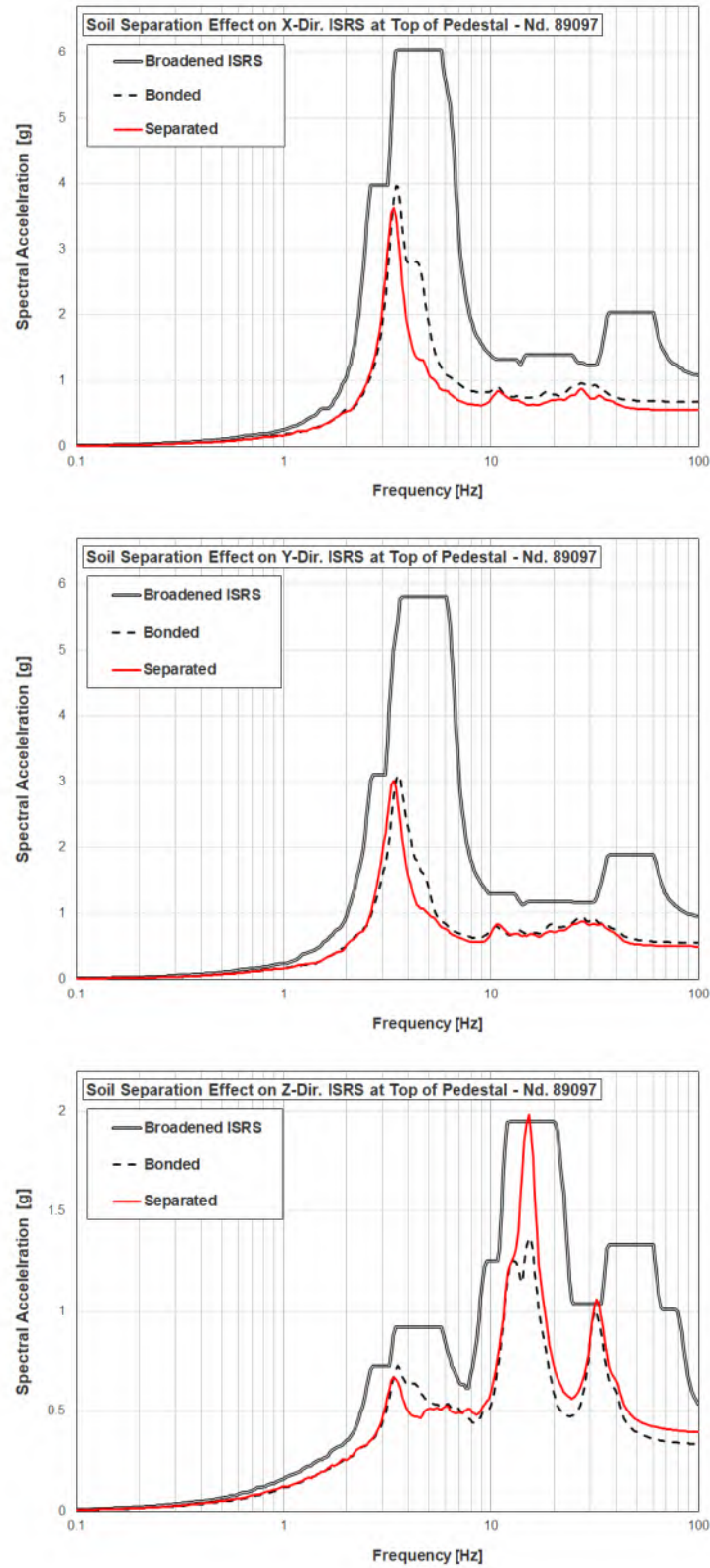


Figure 9B.7-35: Soil Separation Effects on In-Structure Response Spectra for Top of Pedestal

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

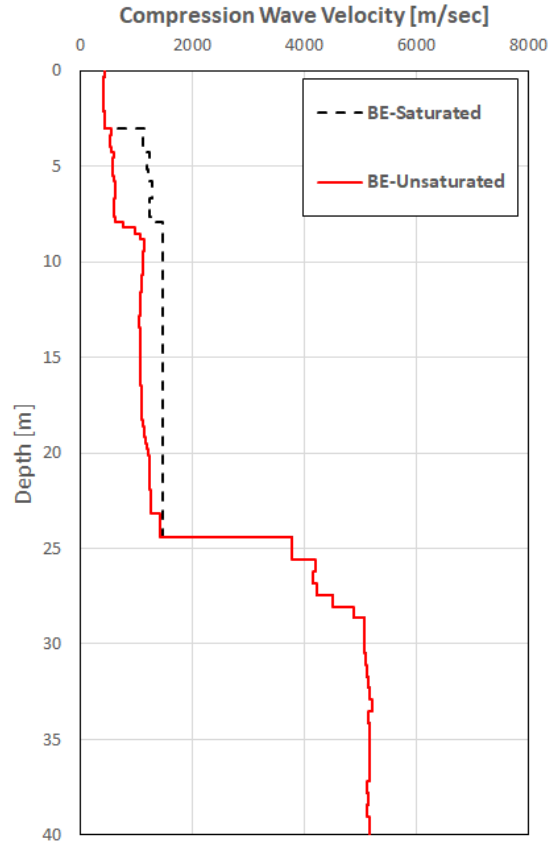
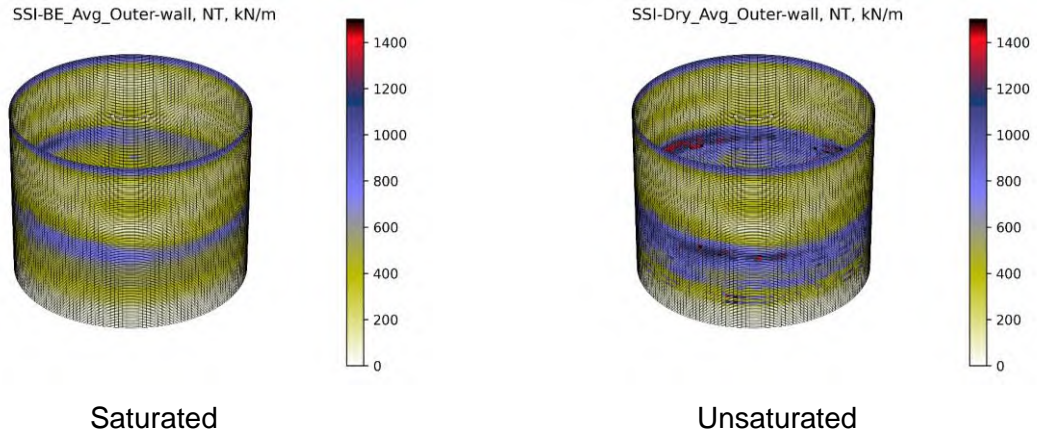


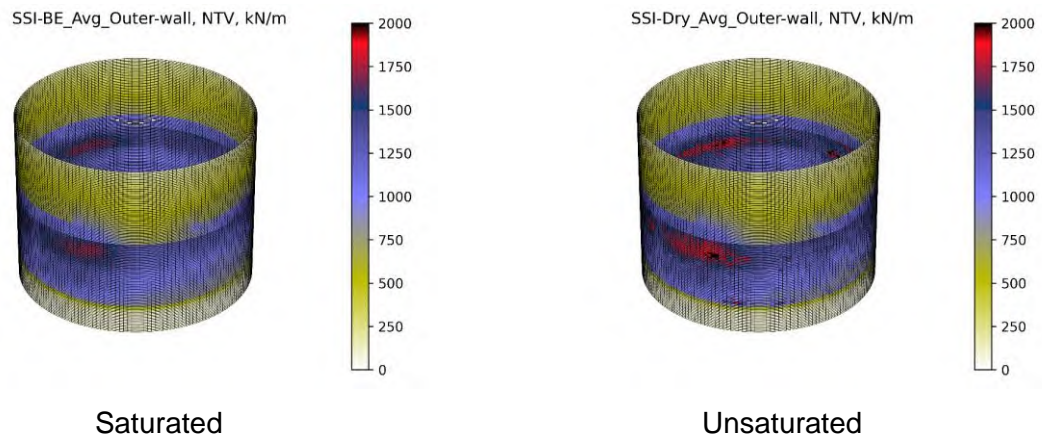
Figure 9B.7-36: Saturated and Unsaturated BE Compression Wave Velocity Profiles



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



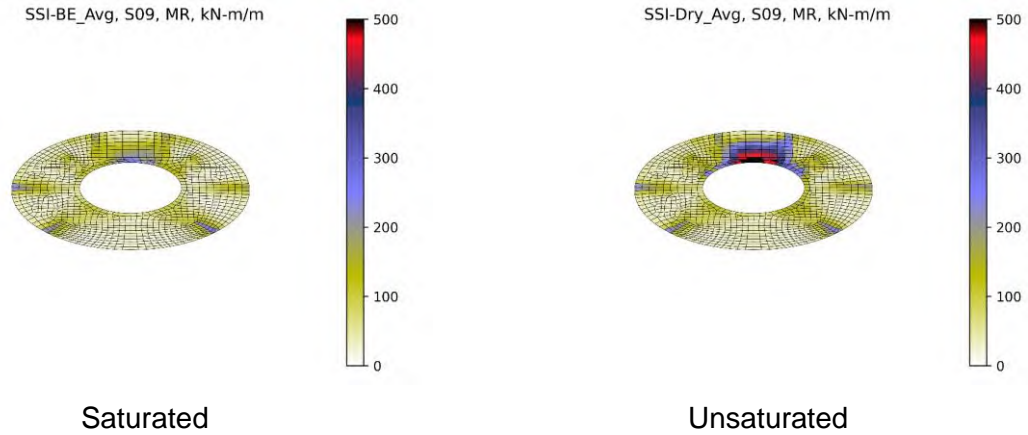
**Figure 9B.7-37: Groundwater Variation Effects on Tangential (Hoop) Forces on the Reactor Building Exterior Wall**



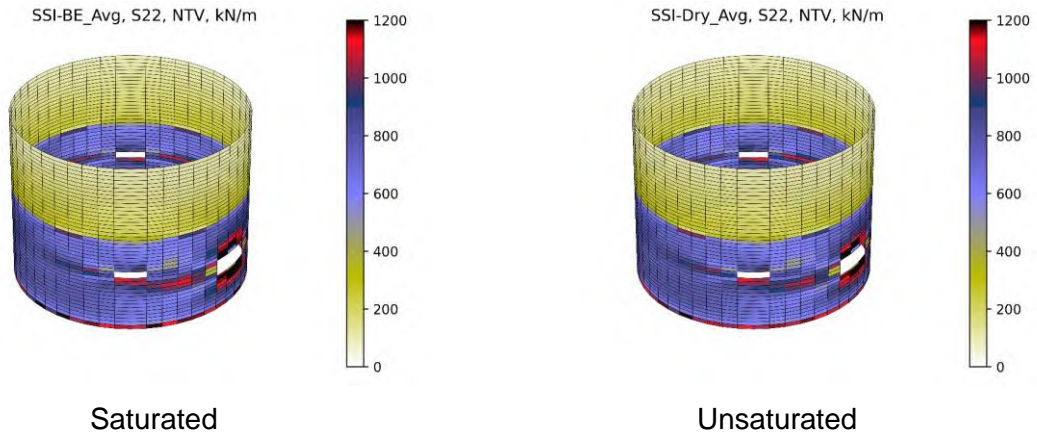
**Figure 9B.7-38: Groundwater Variation Effects on In-Plane Shear in the Reactor Building Exterior Wall**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-39: Groundwater Variation Effects on SCCV Top Slab Bending Moment**



**Figure 9B.7-40: Groundwater Variation Effects on RPV Pedestal In-Plane Shear**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

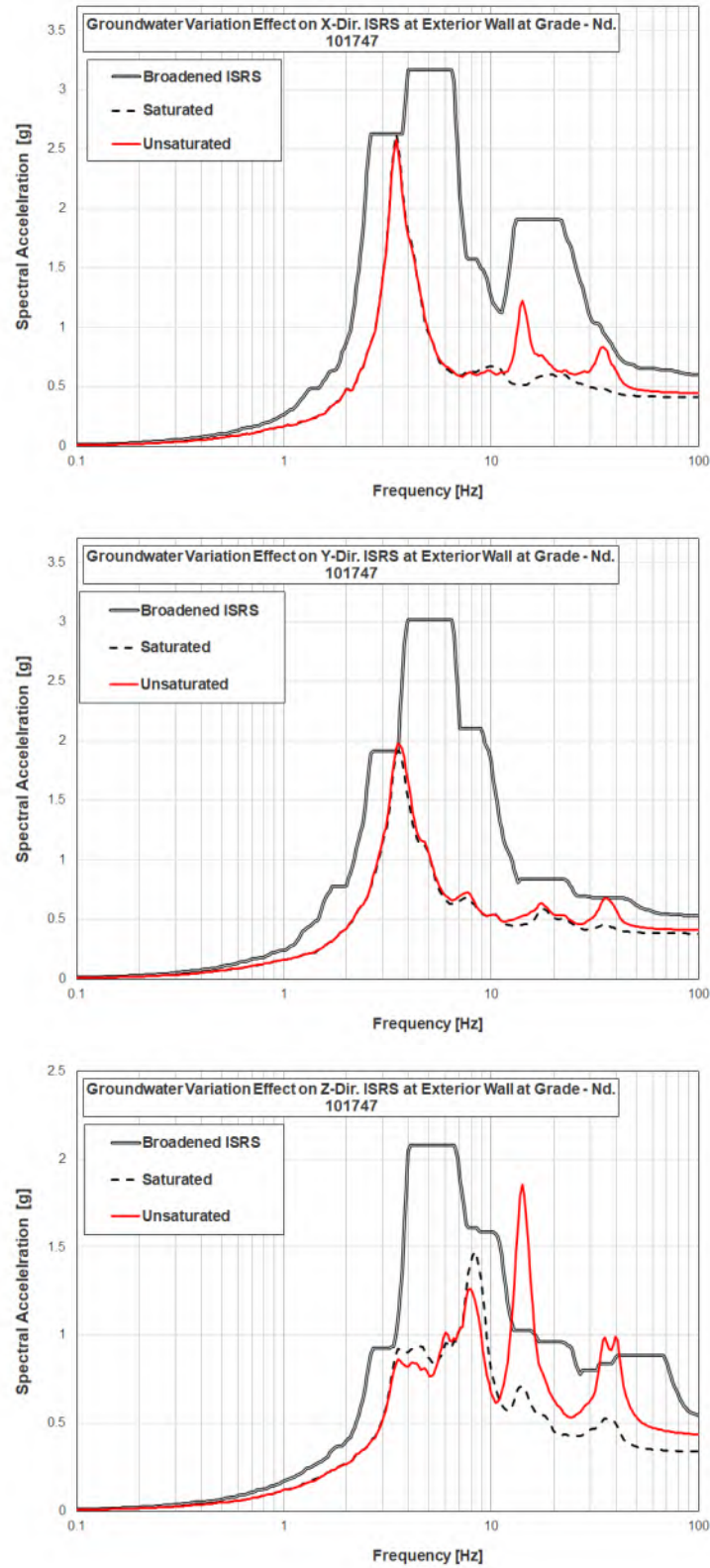


Figure 9B.7-41: Groundwater Variation Effects on In-Structure Response Spectra for Reactor Building Exterior Wall at Grade



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

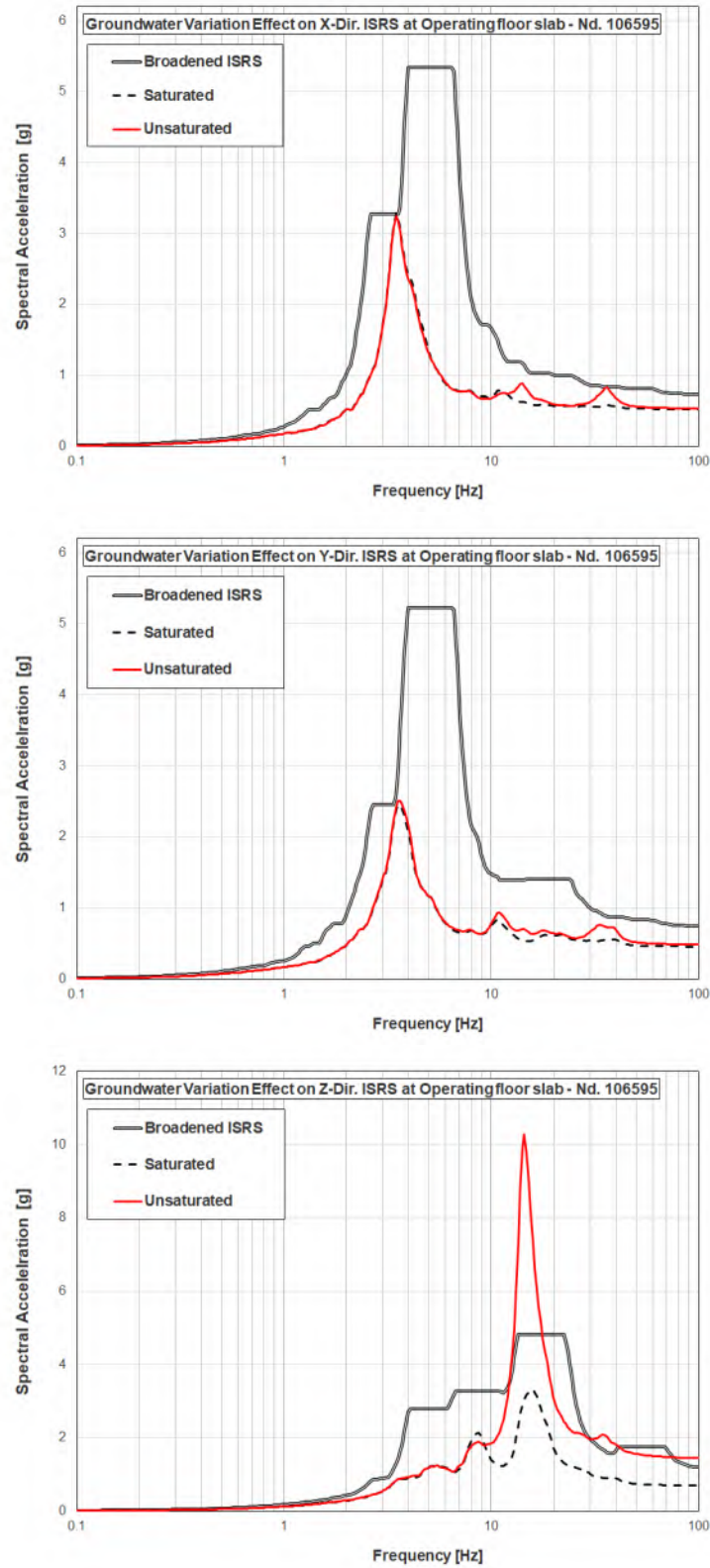
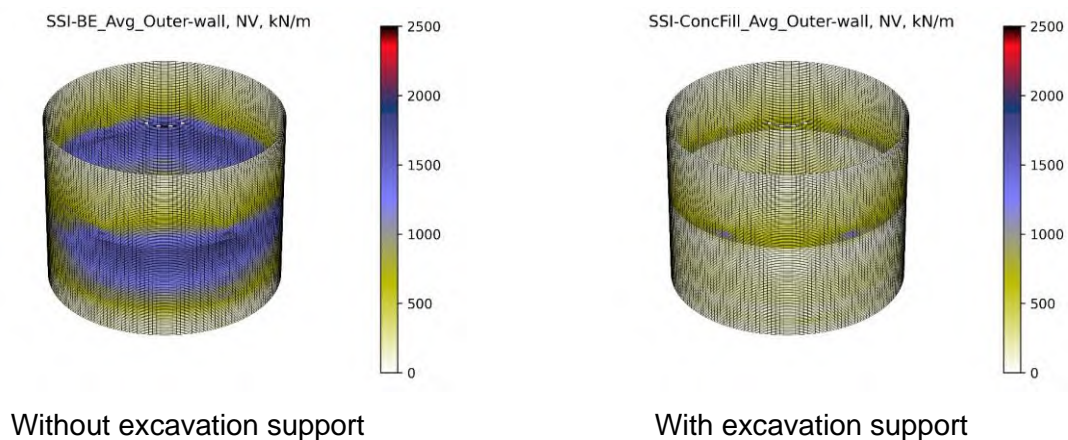
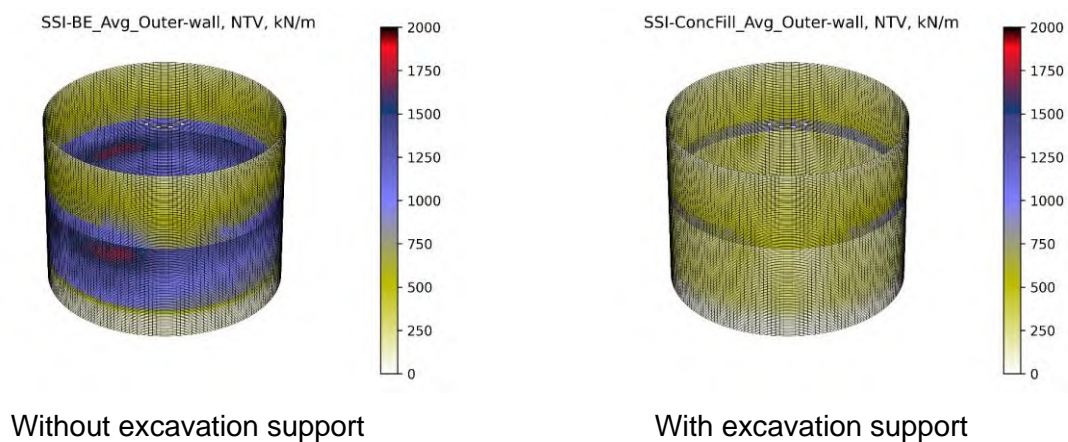


Figure 9B.7-42: Groundwater Variation Effects on In-Structure Response Spectra for Operating Floor Slab

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



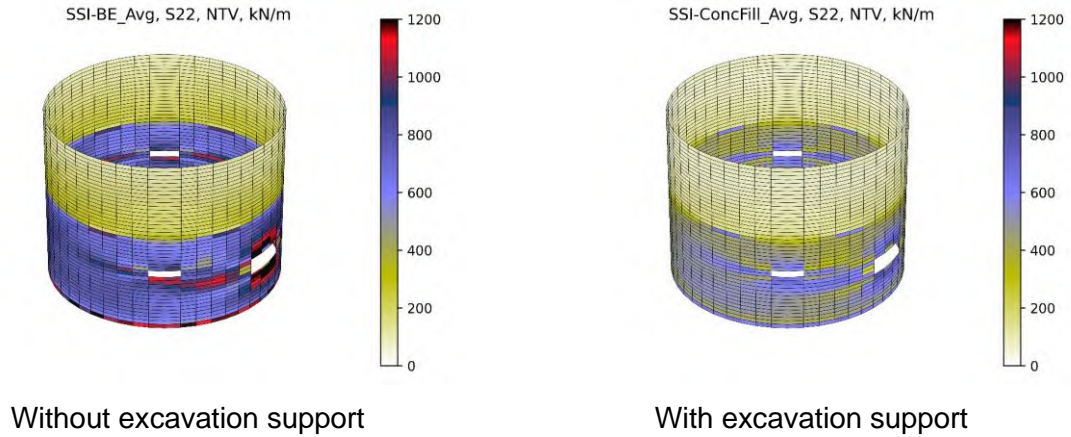
**Figure 9B.7-43: Excavation Support and Backfill Effects on Vertical Forces in the RB Exterior Wall**



**Figure 9B.7-44: Excavation Support and Backfill Effects on In-Plane Shear in the RB Exterior Wall**

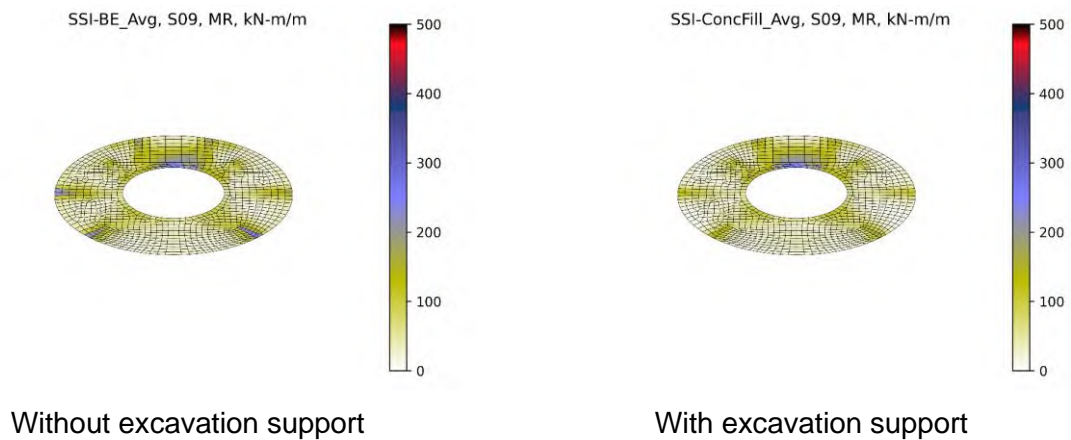
\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



**Figure 9B.7-45: Excavation Support and Backfill Effects on RPV Pedestal In-Plane Shear**

\*Aspect ratio not to scale in these figures



**Figure 9B.7-46: Excavation Support and Backfill Effects on SCCV Top Slab Bending Moment**

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

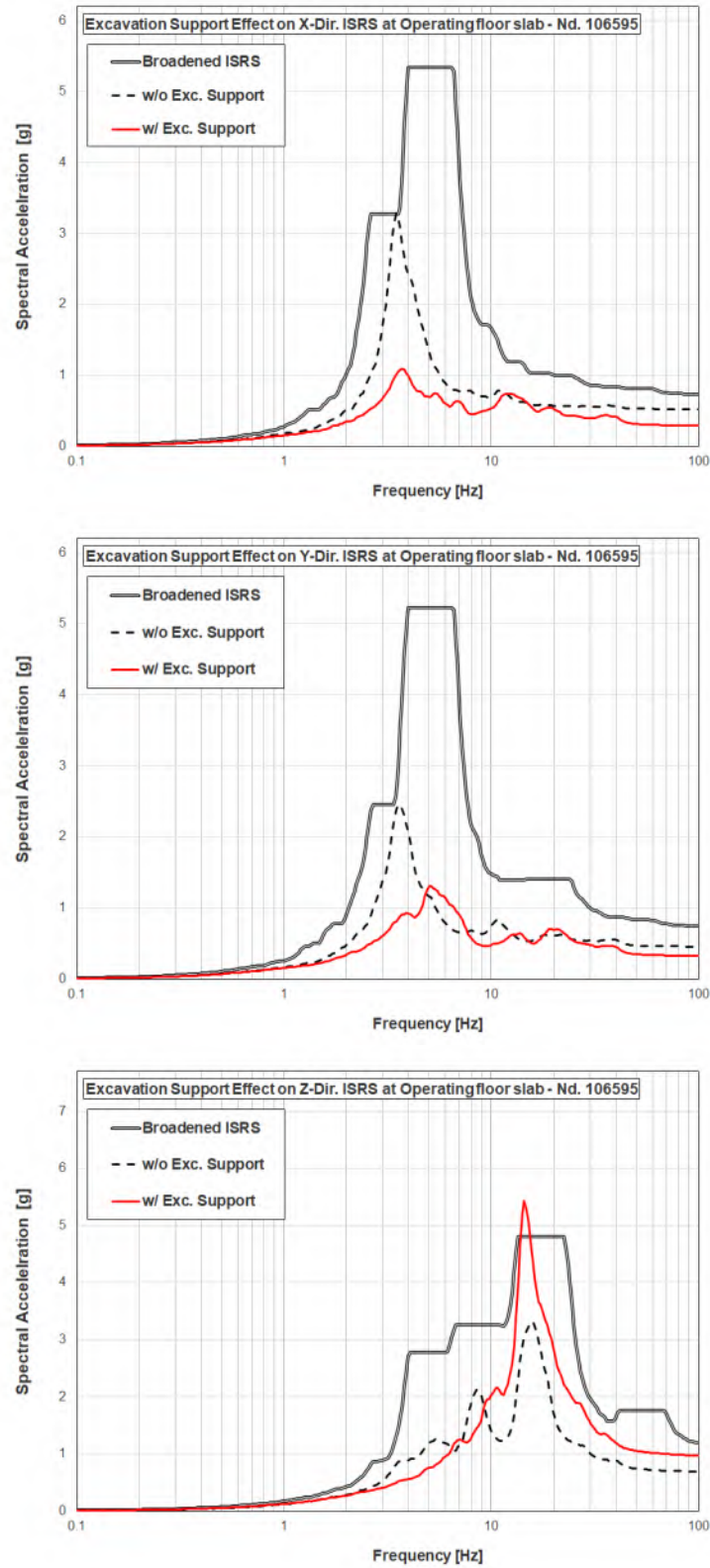


Figure 9B.7-47: Effects of Excavation Support and Backfill on In-Structure Response Spectra for Operating Floor Slab

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

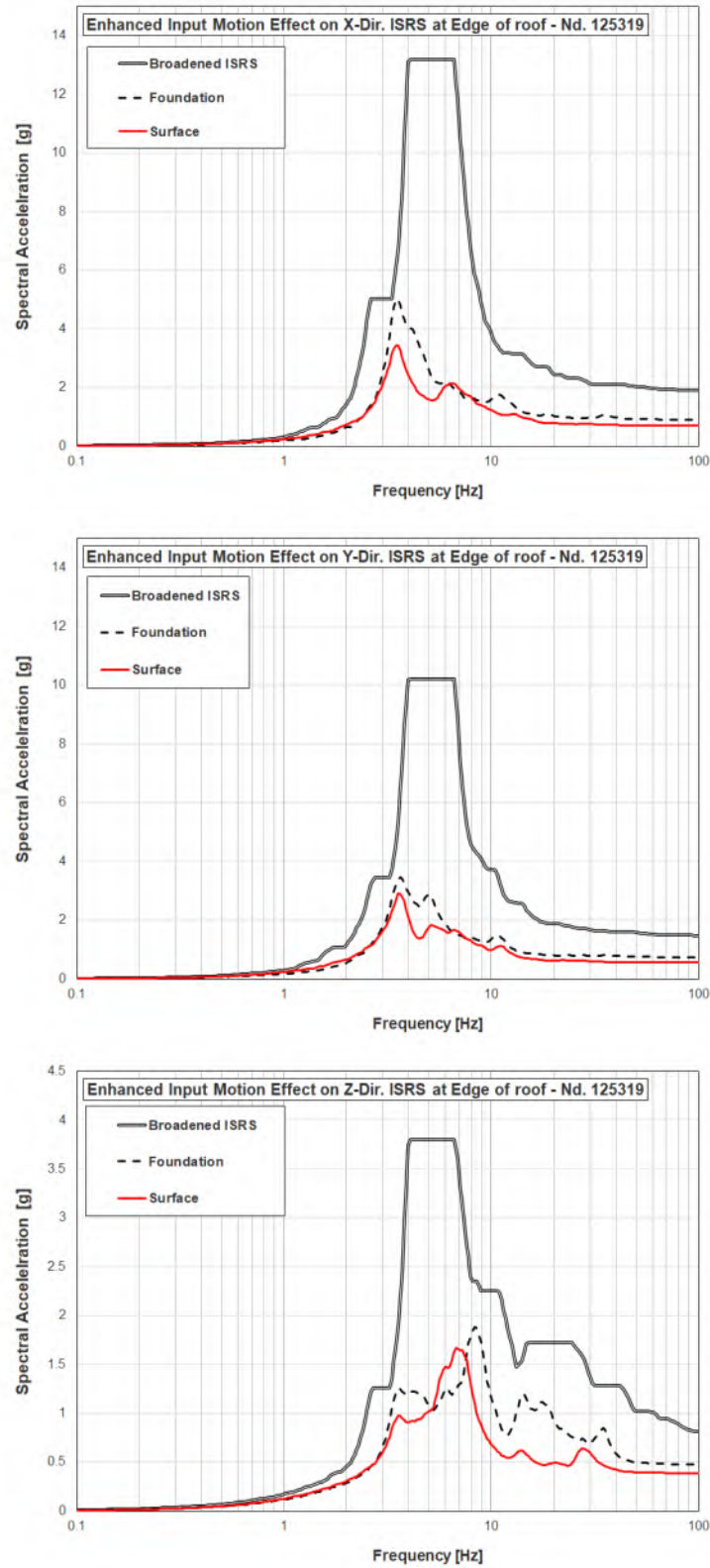


Figure 9B.7-48: Effects of Enhanced Input Motions on In-Structure Response Spectra for Edge of RB Roof



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

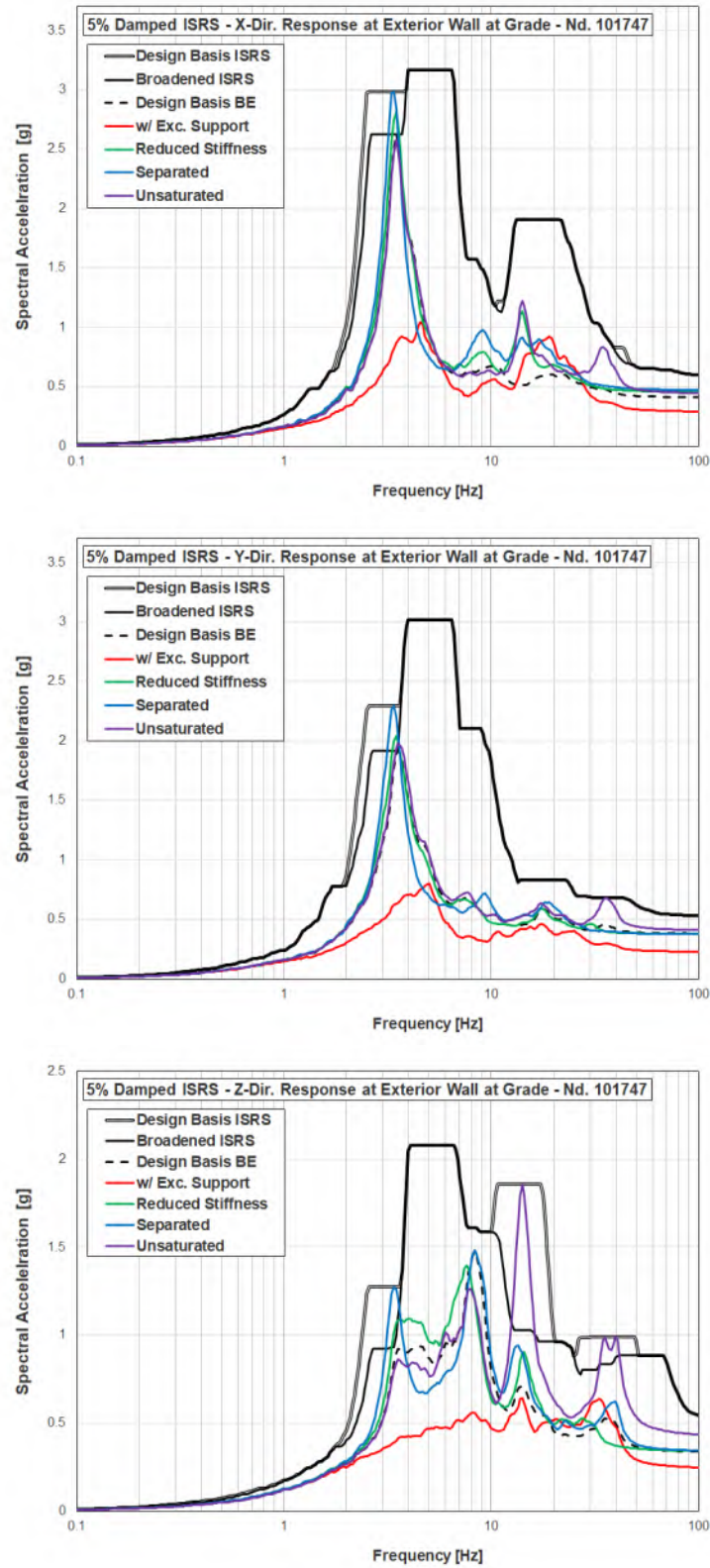
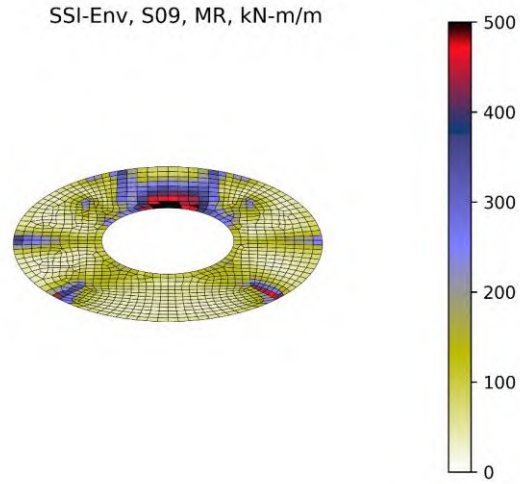
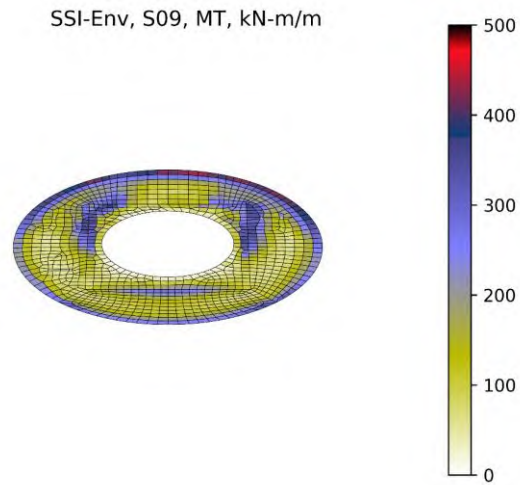


Figure 9B.7-49: SSI Parameter Effects on In-Structure Response Spectra for RB Exterior Wall at Grade

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



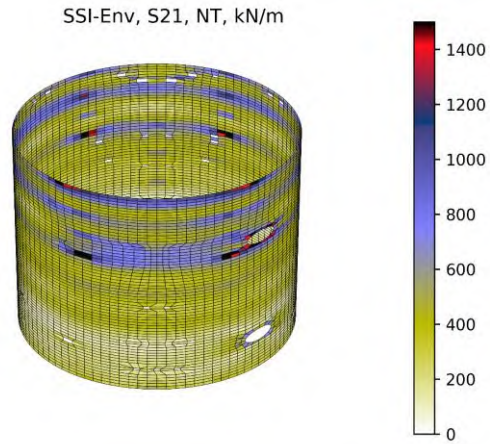
(a) Radial Moment



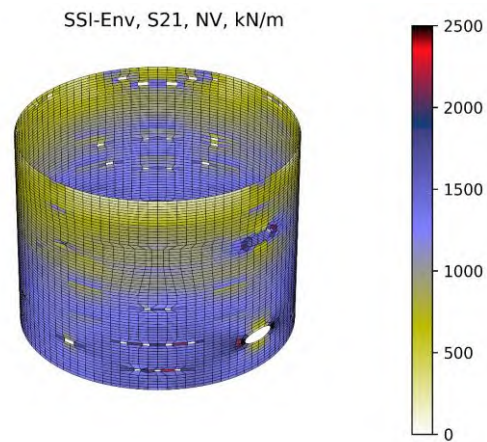
(b) Tangential Moment

**Figure 9B.7-50: Seismic Design Basis Moment Demands for SCCV Top Slab**

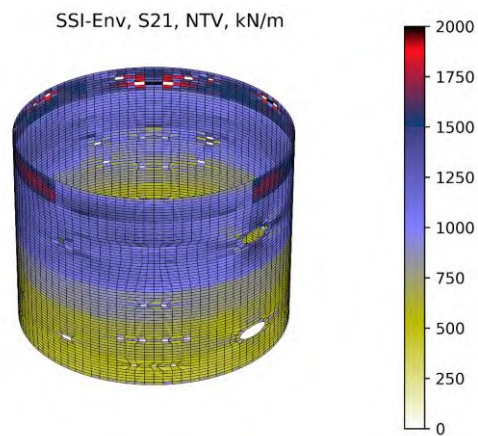
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Tangential (Hoop) Force



(b) Membrane Vertical Force



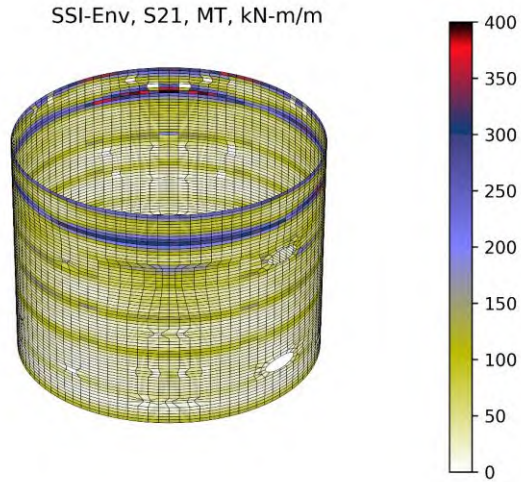
(c) In-Plane Shear

**Figure 9B.7-51: Seismic Design Basis Membrane Forces for SCCV Wall**

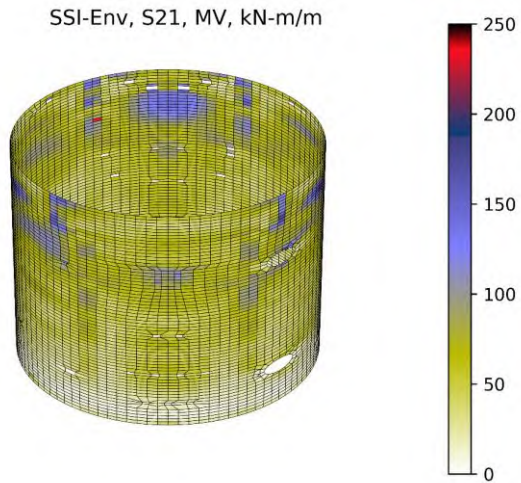
\*Aspect ratio not to scale in these figures



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Tangential Moment

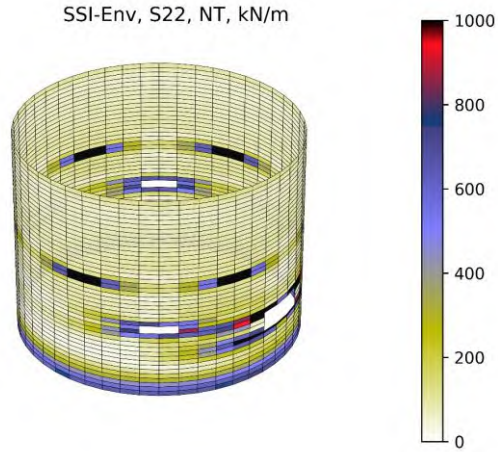


(b) Vertical Moment

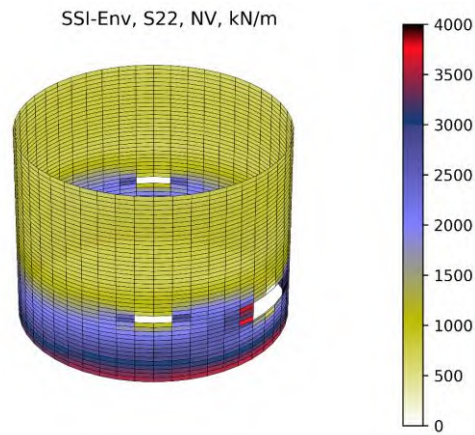
**Figure 9B.7-52: Seismic Design Basis Moment Demands for SCCV Wall**

\*Aspect ratio not to scale in these figures

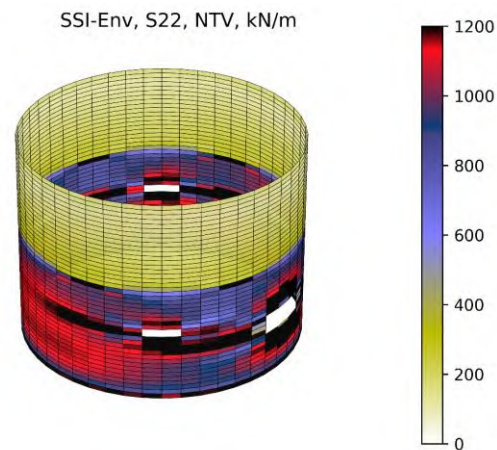
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Tangential (Hoop) Force



(b) Vertical Force

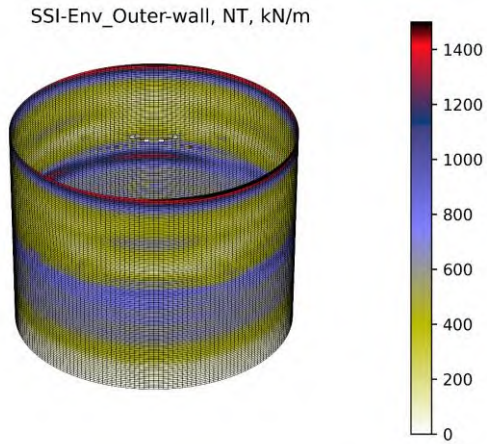


(c) In-Plane Shear

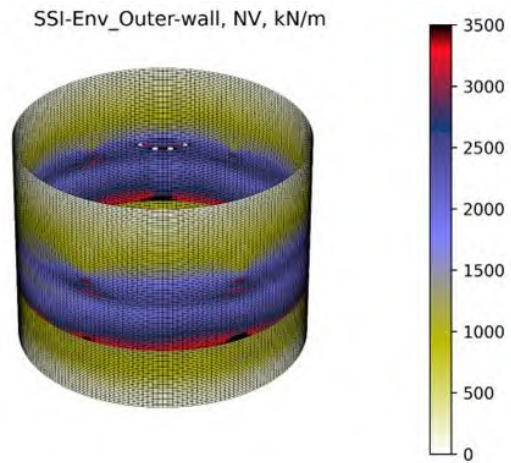
**Figure 9B.7-53: Seismic Design Basis Membrane Forces for RPV Pedestal**

\*Aspect ratio not to scale in these figures

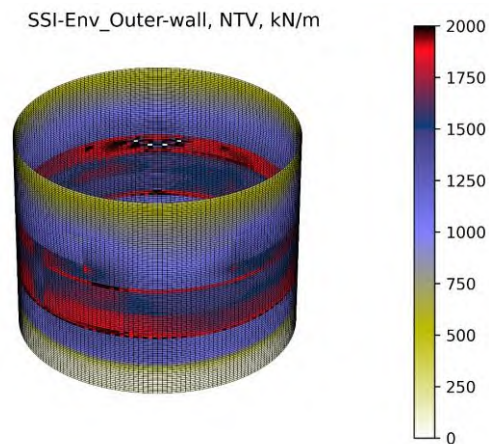
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Tangential (Hoop) Force



(b) Vertical Force

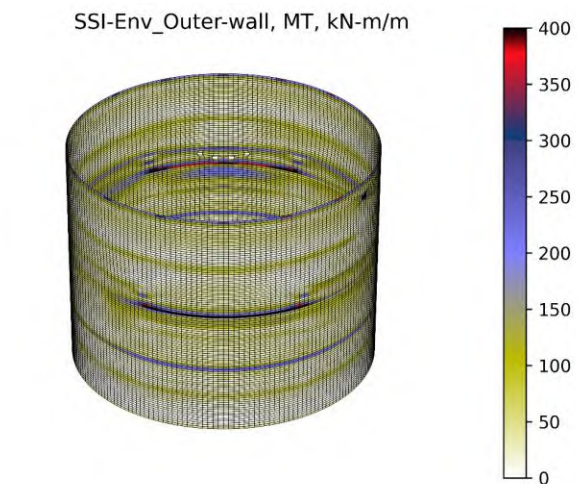


(c) In-Plane Shear

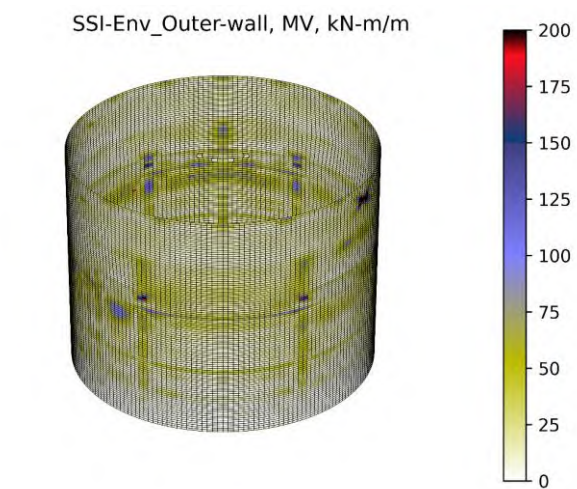
**Figure 9B.7-54: Seismic Design Basis Membrane Forces for Reactor Building Exterior Wall**

\*Aspect ratio not to scale in these figures

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Tangential Bending Moment

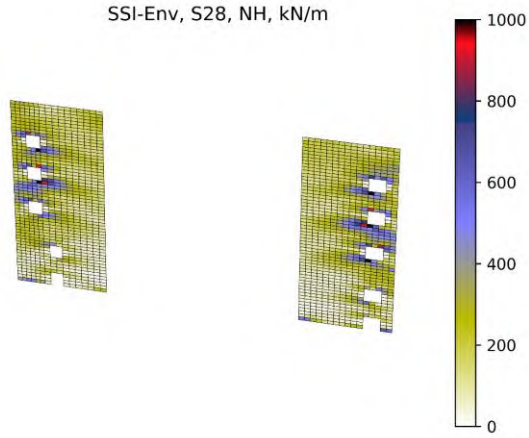


(b) Vertical Bending Moment

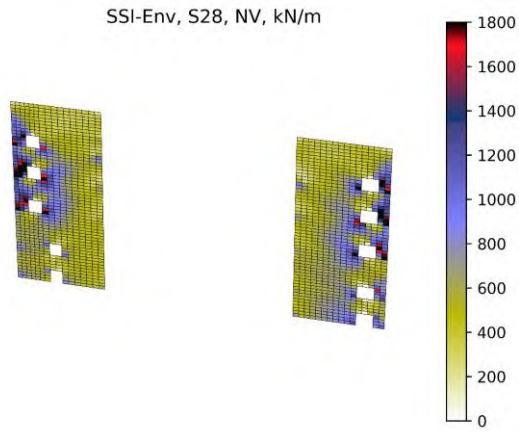
**Figure 9B.7-55: Seismic Design Basis Moment Demands for Reactor Building Exterior Wall**

\*Aspect ratio not to scale in these figures

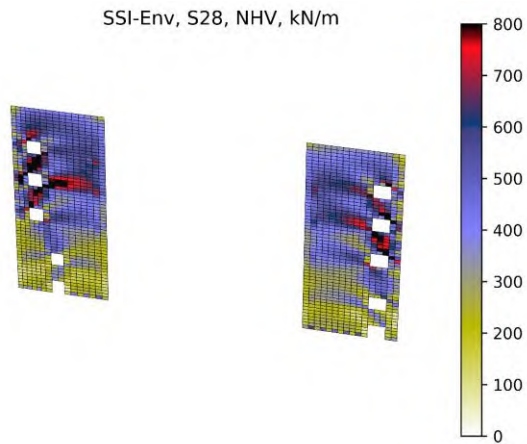
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a) Horizontal Force



(b) Vertical Force



(c) In-Plane Shear

**Figure 9B.7-56: Seismic Design Basis Membrane Forces for Reactor Building Wing Walls**

\*Aspect ratio not to scale in these figures



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

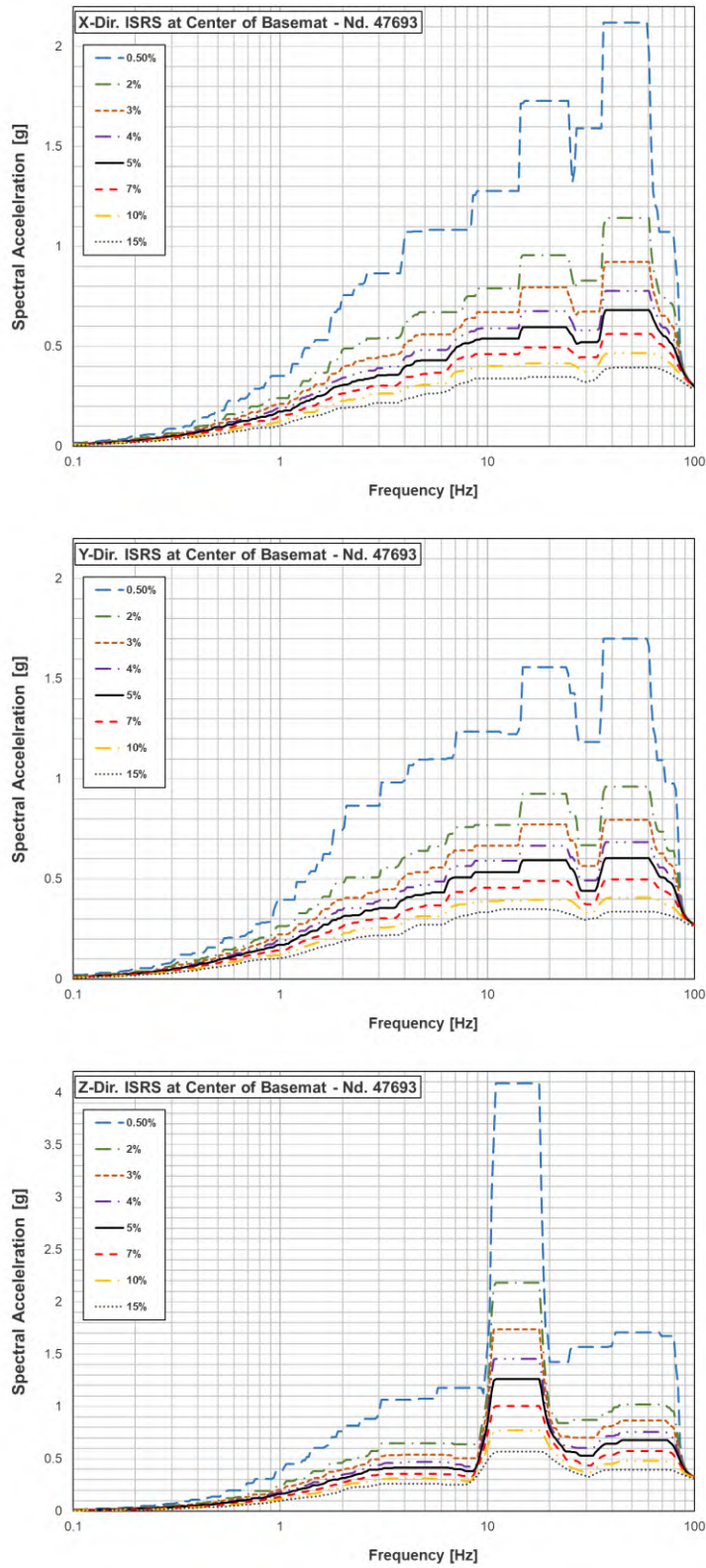


Figure 9B.7-57: Bounding Seismic In-Structure Response Spectra at Centre of Common Mat Foundation

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

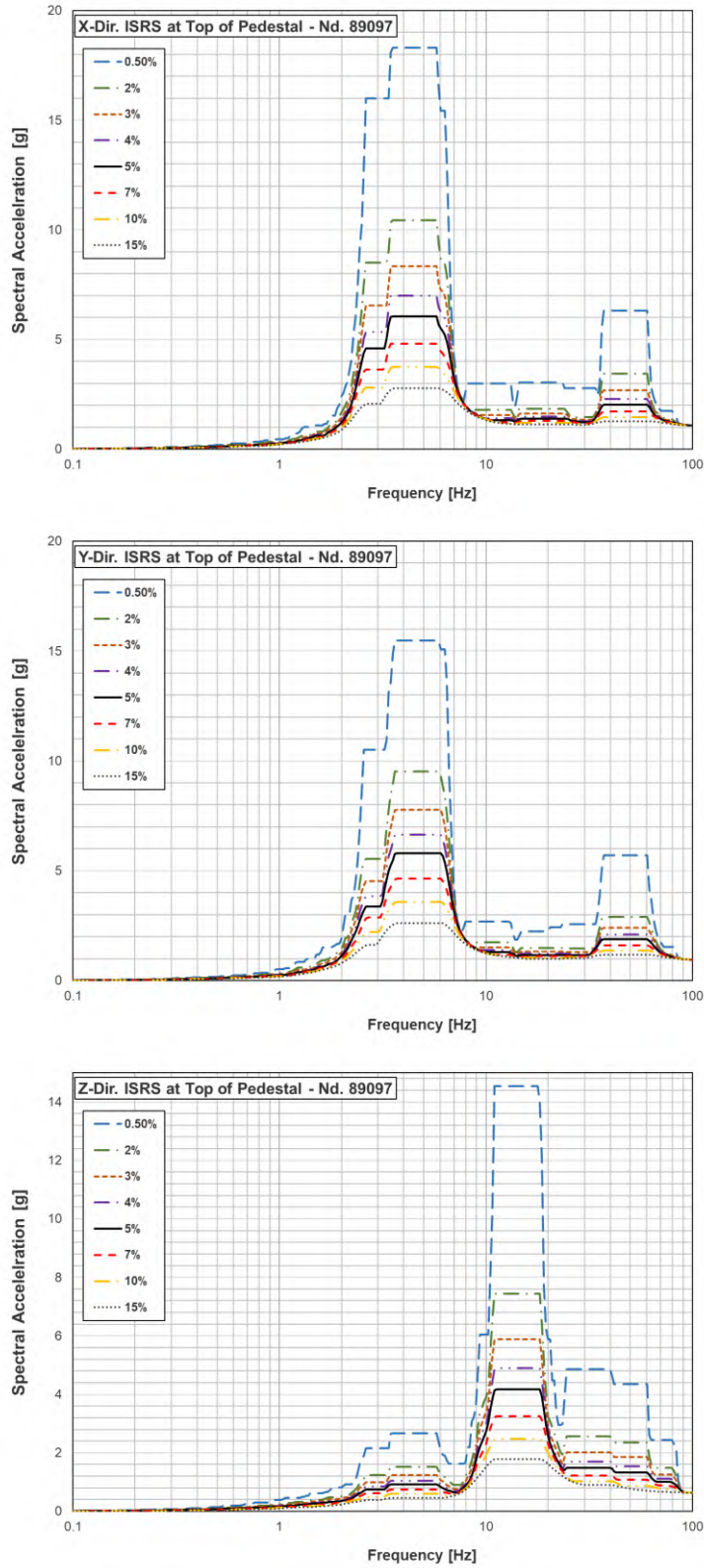


Figure 9B.7-58: Bounding Seismic In-Structure Response Spectra at Top of RPV Pedestal

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

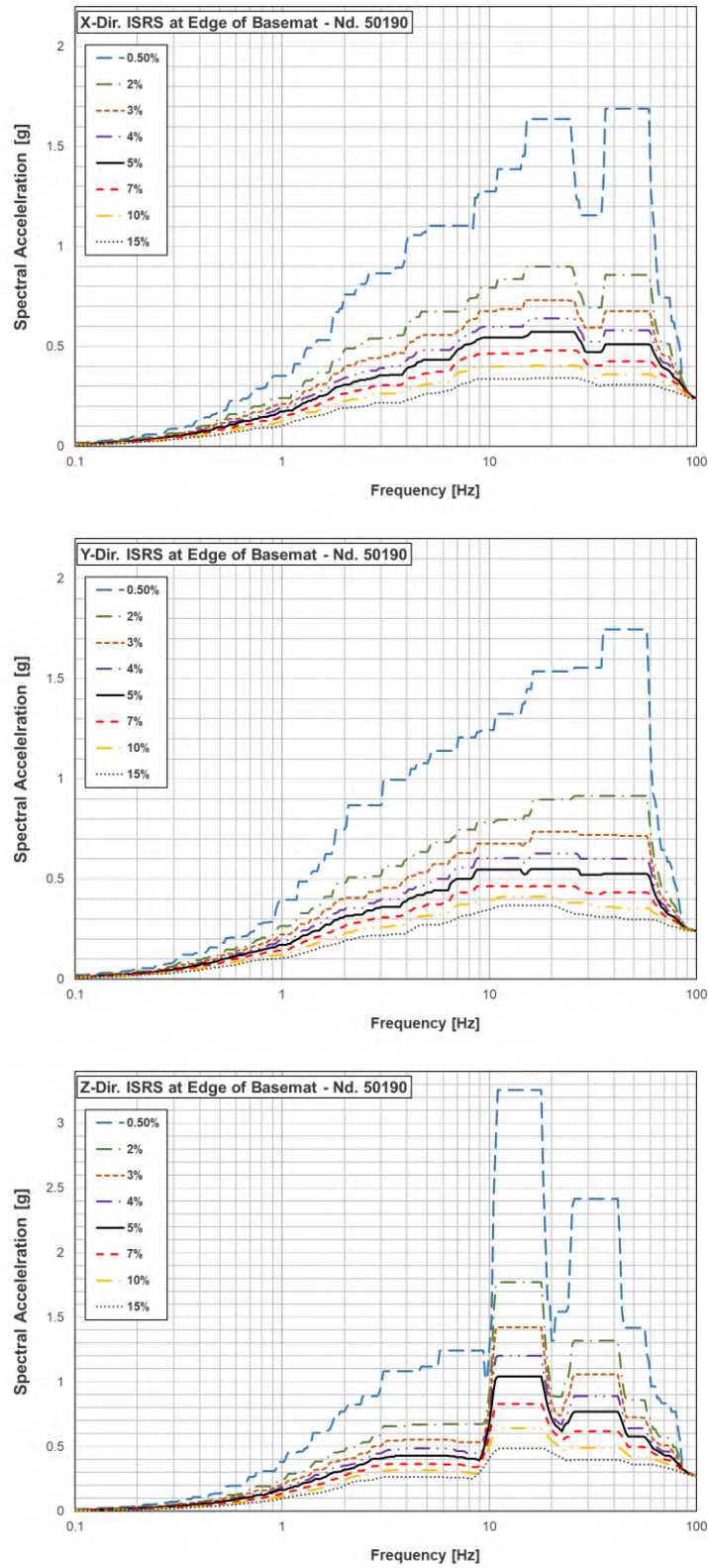


Figure 9B.7-59: Bounding Seismic In-Structure Response Spectra at Edge of Common Mat Foundation



NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

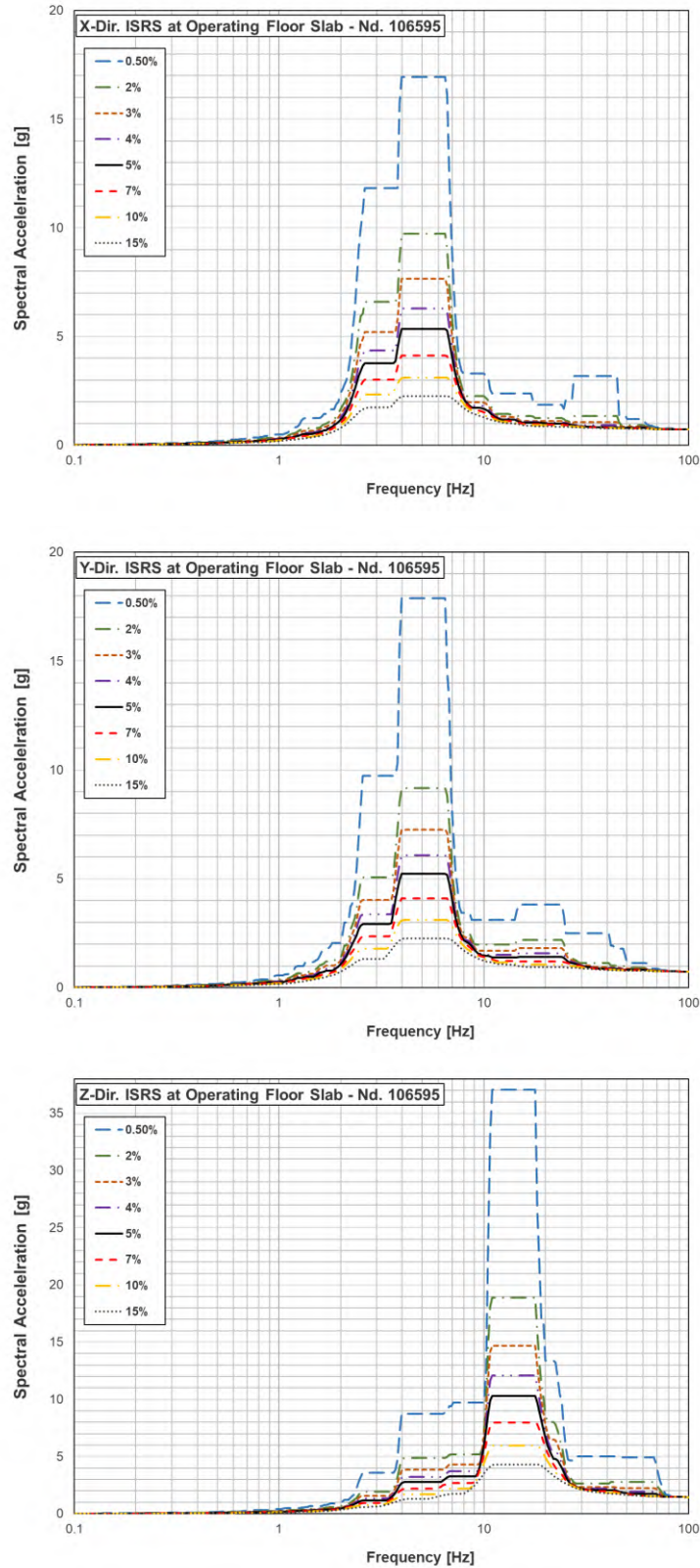


Figure 9B.7-60: Bounding Seismic In-Structure Response Spectra for the Operating Floor Slab

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

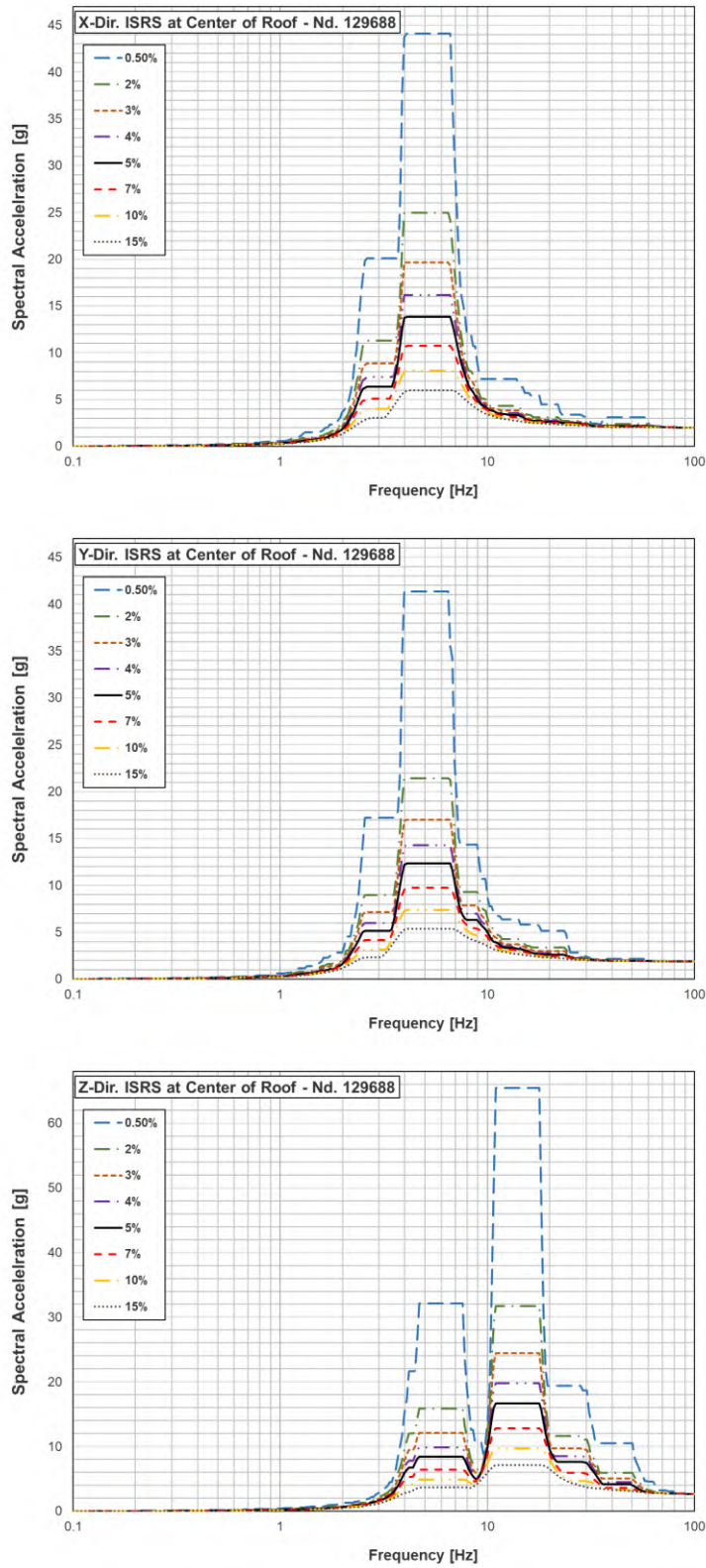


Figure 9B.7-61: Bounding Seismic In-Structure Response Spectra at Centre of RB Roof

## APPENDIX 9B.C – INTERACTION EVALUATIONS RESULTS

### 9B.8 Introduction

The BWRX-300 design prevents the collapse or collision of adjacent Power Block structures and foundations with the RB to prevent impairment of the structural integrity or safety functions of Seismic Category A and B SSCs during and after a DBE or design basis tornado wind events. The RWB, CB, TB and Reactor Auxiliary Bay, referred to in this appendix as the Power Block structures, are all supported by near-surface mat foundations as discussed in Subsection 9B.3.

This appendix provides the seismic interaction evaluations performed to evaluate the structural integrity and stability of the Power Block structures and their foundations during and after a DBE. The evaluations are performed following the guidelines of NEDO-33914, Section 6.2.

#### 9B.8.1 Seismic Interaction Evaluations

##### 9B.8.1.1 Structure-Soil-Structure Interaction Effects

As noted in Appendix 9B.A, Subsection 9B.6.2, the SSSI effects of the Power Block structures on the seismic response of the RB integrated structures are explicitly captured in the bounding DBE demands presented in Appendix 9B.B.

Following the guidance of NEDO-33914, Section 6.2, seismic responses obtained from the design basis analysis of the LB, BE and UB profiles for the coarse dynamic FE models representing the Power Block structures are used to evaluate:

- the seismic sliding and overturning stability of the surface mounted Power Block foundations
- the seismic bearing pressure demands under the surface mounted Power Block foundations and
- the maximum displacement demands for the seismic gaps between the Power Block structures and the neighboring RB exterior wall

##### 9B.8.1.2 Seismic Stability of Power Block Surface Mounted Foundations

The seismic driving forces acting on the Power Block surface mounted foundations are calculated from the results of the contact spring elements at the interfaces of the Power Block foundations and the subgrade. To calculate the co-directional force demands, the time histories of responses due to the three earthquake components are combined algebraically for each element using the time-step-by-time-step approach. The resulting forces are summed up in the time domain for all spring elements under each structure. Maximum resulting driving forces in the three orthogonal directions are recorded for each of the five ATHs for each subgrade profile case and then averaged for the five sets of ATHs.

Seismic driving forces in the three orthogonal directions obtained from the analysis of the LB, BE and UB subgrade profile cases are compared for each of the adjacent Power Block structures to evaluate the effect of subgrade variations. These forces are divided by the corresponding weight of each structure to estimate the effective accelerations at the foundation interface. As demonstrated by Table 9B.8-1 for the TB, the effective accelerations are largest in all three directions for the UB subgrade profile case, followed by the BE and LB cases. This is in line with expectations since base shear forces are associated with high-frequency response. The seismic forces at the base of the Power Block structures, calculated as the envelope for the three subgrade profile cases, are reported in Table 9B.8-2, and the corresponding effective accelerations are reported in Table 9B.8-3.

##### 9B.8.1.2.1 Sliding Stability

Following the methodology in Chapter 3, Subsection 3.3.1.2.10, safety factors for seismic sliding are calculated for each time step, with a minimum safety factor recorded for the duration of each of the

five sets of ATHs. The average value of the minimum safety factors obtained from the five sets of ATHs is used to demonstrate the seismic stability of the structure. The safety factors for sliding stability are calculated using a value of 0.5 for the coefficient of friction between the concrete foundations and the underlying engineered fill.

Table 9B.8-4 presents the results of the sliding stability calculations for the Power Block structures. As shown in Table 9B.8-4, all sliding stability safety factors for the Power Block structures are below the required minimum of 1.1. This indicates that the friction resistance at the base alone is not sufficient to prevent sliding of the near ground surface mounted Power Block foundations. More refined stability evaluations that consider resistance provided by other mechanisms such as foundation side friction, foundation base roughening or shear keys, are to be considered in the next phase of the Power Block surface foundations design.

#### **9B.8.1.2.2 Overturning Stability**

The overturning stability is evaluated by using the energy approach to calculate for each of the Power Block structures a safety factor against overturning. Assuming no soil slip failure occurs, overturning of the structure can be caused only by the centre of gravity of the structure moving far enough horizontally to cause instability.

The mechanism of the rocking motion that may lead to overturning is like an inverted pendulum and its natural period is long compared with the linear, elastic dynamic structural response. Thus, with regard to overturning, the structure can be treated as a rigid body dynamic system for which the maximum kinetic energy is conservatively estimated as follows:

$$E_s = \frac{1}{2}M[V_H^2 + V_V^2]$$

Where: M is the total mass of the structure

$V_H$  and  $V_V$  are the maximum horizontal and vertical velocities, respectively

To simplify calculations,  $V_H$  and  $V_V$  are conservatively calculated from the results of the SSI analyses for the responses at the roof elevation of the structure. The velocity in each of the three orthogonal directions is calculated at each time step of the ATH as the summation of the contribution from each of the three orthogonal input motions. The maximum absolute value for each ATH is recorded and then averaged over the five sets of ATHs. The horizontal velocity  $V_H$  is considered the maximum of the two orthogonal velocities,  $V_X$  and  $V_Y$ .

Given that the bottom elevations of all Power Block foundations are located above the nominal groundwater table elevation, the effect of buoyancy is immaterial, and the resisting energy is calculated as:

$$E_o = Mgh$$

Where: g is the acceleration of gravity

h is the height to which the centre of the structure has to be lifted to reach the overturning position

The overturning height h is estimated as follows:

$$h = h_o - h_s$$

Where:  $h_s$  is the height of centre of mass of the structure

$h_o$  is the SRSS of  $h_s$  and half the width of the foundation footprint.

The safety factor against overturning is calculated as the ratio of the resisting energy  $E_o$  to the maximum kinetic energy  $E_s$  and is required to be larger than 1.1. The calculation is performed for each of the subgrade profile cases and the lowest safety factor is recorded for each structure. Table 9B.8-5 presents the resulting safety factors against overturning for all Power Block structures. As shown in Table 9B.8-5, the computed safety factors are all much larger than the required safety factor of 1.1. This indicates that overturning instability is not a critical failure mode for any of the Power Block structures.

#### **9B.8.1.3 Seismic Bearing Pressure Demands for Surface Mounted Power Block Foundations**

Table 9B.8-6 presents the average seismic bearing pressures under each of the Power Block near-surface mounted foundations. The seismic bearing pressures are calculated as the vertical dynamic force at the base of each structure, reported in Table 9B.8-2, divided by the foundation footprint area.

As shown in Table 9B.8-6, the largest pressure observed is under the RWB foundation and is 54.1 kPa. This pressure is within the allowable serviceability and factored limit states listed in Chapter 2, Subsection 2.7.5.1.

#### **9B.8.1.4 Relative Displacement**

Seismic horizontal relative displacements are calculated for each of the Power Block structures relative to the nearby RB exterior wall using results from the LB, BE and UB subgrade profile cases. Relative displacements are computed between a node on the roof of the considered structure and the nearest node of the RB exterior wall.

For each set of ATHs, the maximum relative displacement obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response using the SRSS method. For each subgrade profile analysis, average relative displacements from the five sets of ATHs are computed representing the maximum relative displacement for the considered subgrade profile case.

As demonstrated by Table 9B.8-7 presenting the maximum relative displacements between the RB and TB, the analysis of the LB subgrade profile case provides the largest relative displacements in all three directions followed by the results of the BE and UB profiles.

Table 9B.8-8 presents the maximum seismic relative displacements computed for the Power Block structures relative to the RB structure. As noted in this table, the seismic gaps between the RB exterior wall and the stiff RWB and CB reinforced concrete shear wall structures are adequate to accommodate the DBE generated relative displacements, which based on the bounding SSI analysis results are less than 21 mm. The large maximum relative displacements of 74 mm and 53 mm listed in Table 9B.8-8 for the TB and Reactor Auxiliary Bay, respectively, are primarily due to the deformation of these flexible structures. These relative displacements indicate the need for increasing the lateral stiffness of the TB and Reactor Auxiliary Bay structures in the next phase of the design to prevent the closing of the seismic gap between these buildings and the RB.

**Table 9B.8-1: Subgrade Variation Effects on Seismic Driving Forces**

Subgrade Profile Case	Acceleration in X direction (North-South) (g)	Acceleration in Y direction (East-West) (g)	Acceleration in Z direction (Vertical) (g)
LB	0.34	0.34	0.28
BE	0.42	0.38	0.32
UB	<b>0.47</b>	<b>0.39</b>	<b>0.34</b>

\* Values presented in this table are for the TB and are for illustration purposes only

**Table 9B.8-2: DBE Seismic Forces at the Base of Power Block Structures**

Power Block Structure	Base Shear Force in X direction (North-South) (kN)	Base Shear Force in direction (East-West) (kN)	Base Force in Z direction (Vertical) (kN)
RWB	111,056	110,155	57,677
TB	271,928	224,023	198,236
CB	83,158	78,254	45,848
Reactor Auxiliary Bay	6,058	7,152	6,181

**Table 9B.8-3: DBE Effective Acceleration Demands at the Base of Power Block Structures**

Power Block Structure	Acceleration in X direction (North-South) (g)	Acceleration in Y direction (East-West) (g)	Acceleration in Z direction (Vertical) (g)
RWB	0.73	0.72	0.38
TB	0.47	0.39	0.34
CB	0.72	0.68	0.40
Reactor Auxiliary Bay	0.50	0.58	0.51

**Table 9B.8-4: DBE Sliding Stability for Power Block Structures**

Power Block Structure	Sliding Safety Factor
RWB	0.56
TB	0.94
CB	0.55
Reactor Auxiliary Bay	0.66

**Table 9B.8-5: DBE Overturning Stability for Power Block Structures**

Power Block Structure	$E_O$ (kN-m)	$V_H$ (m/sec)	$V_V$ (m/sec)	$E_S$ (kN-m)	Overturning Safety Factor
RWB	891,145	0.47	0.16	2,828	315
TB	13,715,113	0.95	0.21	23,037	595
CB	1,584,390	0.27	0.16	572	2,771
Reactor Auxiliary Bay	27,342	1.01	0.15	654	42

**Table 9B.8-6: DBE Seismic Bearing Stress at the Base of Power Block Structures**

Power Block Structure	Seismic Bearing Pressure (kPa)
RWB	54.1
TB	41.2
CB	19.4
RAB	14.6

**Table 9B.8-7: Maximum Relative Displacements between the Reactor Building and Turbine Building**

Subgrade Profile	Displacement in X direction (North-South) (mm)	Displacement in Y direction (East-West) (mm)	Displacement in Z direction (Vertical) (mm)
LB	74.1	64.4	16.7
BE	50.6	51.0	7.7
UB	32.6	26.6	5.2

**Table 9B.8-8: DBE Maximum Relative Displacements between the Reactor Building and Surrounding Power Block Structures**

Subgrade Profile Case	Displacement in X direction (North-South) (mm)	Displacement in Y direction (East-West) (mm)	Displacement in Z direction (Vertical) (mm)
RWB	17.9	20.7	11.3
TB	74.1	64.4	16.7
CB	9.7	8.5	4.1
RAB	52.7	50.6	9.0



## APPENDIX 9B.D – RESPONSES TO STATIC LOADS

### 9B.9 Introduction

This appendix documents the 1-g static SSI, static and quasi-static and thermal analyses performed for the design of the integrated RB structures. The analyses are performed per the methodology presented in Chapter 3, Subsection 3.5.1.1, using the FE models described in Appendix 9B.A, Subsections 9B.6.3 through 9B.6.5, and design parameters discussed in Subsection 9B.9.1.

The 1-g static SSI, static and quasi-static and thermal analyses are performed for the applicable loads listed in Chapter 3, Subsection 3.5, excluding the crane payload, tornado missiles, flooding, accidental loading outside of containment, and piping reaction loading for which information is not available. These loads are judged to be non-governing for the bounding design of the integrated RB structures and will be considered in the final design of the structures.

#### 9B.9.1 Design Parameters Considered in the Structural Analyses

##### 9B.9.1.1 Bounding Equivalent Linear Static Subgrade Properties

The bounding site-specific geotechnical parameters presented in Chapter 3, Table 3.5-1 and Table 3.5-2 are used to develop input for the 1-g static SSI and subgrade impedance analyses used to evaluate the structural response to significant static and thermal design loads. The use of the bounding parameters results in force/moment demands that conservatively envelop those obtained using the geotechnical properties of the subsurface materials at the DNNP site.

Table 9B.9-1 identifies the equivalent linear static subgrade profiles used for each of the structural analyses. The LB and UB elastic moduli ( $E_{st}$ ) in Table 9B.9-1 represent the lower and upper bound moduli from as-built static subgrade properties in Chapter 3, Table 3.5-1 and Table 3.5-2. The range of at-rest lateral earth pressure coefficients are used to calculate the upper soil layer's UB and LB static Poisson ratio's, respectively. The effective (submerged) soil unit weights in Table 9B.9-1 are based on UB soil weight estimates calculated from the average soil weights in Chapter 3, Tables 3.5-1 and Table 3.5-2.

LB  $E_{st}$  stiffness properties are used for the 1-g SSI, static and quasi-static analyses to emphasize the deformations at the subgrade-structure interfaces resulting in UB estimates of member forces and moment demands. LB Poisson ratios are used for the static and quasi-static analyses to emphasize the lateral deformations of the RB integrated structures under mechanical design loads. The 1-g SSI analyses use UB Poisson ratios to emphasize the lateral earth pressures applied from the subgrade on the RB exterior wall. UB subgrade stiffness properties UB  $E_{st}$  and UB  $\nu_{st}$  are considered for the thermal stress analyses to emphasize the restraint from the subgrade against the thermal expansion of the containment resulting in UB thermal stress demands on the RB integrated structures.

##### 9B.9.1.2 Nominal Groundwater Table Elevation

The nominal water table depth considered in the bounding design of the integrated RB at the DNNP site is set at 3 m below grade as stated in Chapter 3, Subsection 3.5.2.2.

##### 9B.9.1.3 In-Situ Residual Rock Pressure

Residual horizontal stresses with magnitudes that exceed the magnitude of the vertical stresses may be present in the rock masses at the DNNP site that are due to the past seismic activities. Results of past studies indicate that these stresses are anisotropic in nature, i.e., with different pressure magnitudes in the two orthogonal horizontal directions.

Rock excavation can relieve the residual horizontal stresses due to deformation of the rock mass. However, the magnitude of the rock mass displacements and resulting relief of residual rock stresses depends on the duration of the excavation and rock reinforcement that may be used to secure the

excavation. The durability of the temporary rock reinforcement elements during the life of the plant is not guaranteed. To account for possible decay of rock reinforcement with time, horizontal in-situ residual rock stresses are applied on the RB exterior wall.

The bounding residual rock pressures without consideration for relief during excavation are estimated using the Poisson's ratios for the DNNP site that include (0.43) and exclude (0.30) in-situ residual stresses. The design of the RB integrated structure conservatively considers that 50% or less of the bounding residual in-situ stresses can conservatively account for the residual rock pressure that may be applied on the RB exterior wall during the operational life of the BWRX-300.

Lateral compression pressures applied to the model to account for these residual rock pressures are:

- 350 kPa in the N70E and S70W directions
- 250 kPa in the N20W and S20E directions

These loads are applied uniformly along the depth of the embedded RB exterior wall starting at depth of 23.9 m. The magnitude of the external rock pressure at any other direction is computed using linear interpolation between these two values.

#### **9B.9.1.4 Ambient Temperature**

The stress-free temperature for the design of the RB and the SCCV is 15.5°C. Per Chapter 2, Table 2.6-1, ambient design temperatures considered in the thermal analyses are -40°C in winter and 40°C in summer.

#### **9B.9.1.5 Precipitation Design Parameters**

The site-specific design of the DNNP RB uses normal and extreme ground snow loading of 2.5 kPa and 5.0 kPa, respectively. These values bound the ground snow loading at the DNNP site as discussed in Chapter 3, Subsection 3.3.2.3.

#### **9B.9.1.6 Wind and Tornado Parameters**

The integrated RB is designed for a wind load based on a 257.5 km/h 3-sec gust wind, which bounds the 3-sec gust wind for the DNNP site as discussed in Chapter 3, Subsection 3.3.2.4.

The design basis tornado wind and tornado missiles used in the design of the integrated RB are listed in Table 9B.9-2 and Table 9B.9-3, respectively. These parameters are based on Region I parameters listed in Tables 1 and 2 of US NRC Guide 1.76 (Reference 9B.4-51) and envelop those of the design basis tornado for the DNNP site presented in Chapter 2, Subsection 2.6.6, categorized as equivalent to Region-II of U.S. NRC RG 1.76.

The current tornado missile analysis does not assess the impact of an automobile striking the RB from the east side or the impact of rigid missiles (Schedule 40 pipe and 1" solid sphere) on RB walls and roof thicknesses. These assessments are considered to have minor effects on the current design of the integrated RB and will be considered in the final design of the structure.

#### **9B.9.1.7 Pool Water Levels**

Pool water levels used for the hydrostatic load case are based on normal operating levels and are as follows:

- 7.2 m for the ICS Pool
- 7.8 m for the equipment and reactor cavity pools
- 12.7 m for the fuel pool

#### **9B.9.1.8 Containment Internal Pressure Loading**

Containment pressure loading considered in the analyses are presented in Table 9B.9-4.

Conservatively, the negative containment pressure which does not occur under normal operation but can occur during AOO is considered for the normal operating load case.

Test pressure load ( $P_t$ ) applied to the SCCV is defined as 1.15 times the containment design pressure as stated in Subsection 9B.2.1.8.

#### **9B.9.1.9 Thermal Conditions**

Table 9B.9-5 and Table 9B.9-6 summarize the design winter and summer thermal conditions inside and outside the integrated RB during normal operations and DBAs, respectively, considered in the analyses.

#### **9B.9.2 1-g Static SSI Analysis**

The 1-g SSI analysis is performed using the ACS SASSI (See Chapter 3, Appendix 3B) flexible volume sub-structuring method to consider the interaction of the deeply embedded RB with the surrounding Power Block structures and subgrade under 1-g loads. The model used for the 1-g SSI analyses is described in Appendix 9B.A, Subsection 9B.6.3. The soil and rock layers of the far-field, near-field and excavated volumes of the model are assigned the equivalent linear elastic subgrade properties summarized in Table 9B.9-1 and provided in Chapter 3, Table 3.5-1 and Table 3.5-2. Structural elements of the model are assigned BE stiffness properties as discussed in Appendix 9B.A, Subsection 9B.6.3. Submerged unit weight properties are assigned to the subgrade materials located below the bounding nominal groundwater level specified at depth of 3 m below the plant grade as noted in Subsection 9B.1.2.

The subtraction method approximation is used for the 1-g SSI analysis with interaction nodes being specified only at the interface of the excavated volume with the far-field. The use of the subtraction method approximation for the 1-g SSI analysis is adequate since the analyses consider the response at very low frequencies.

As described in Chapter 3, Subsection 3.3.1.2.1, the results of different sub-analyses, are combined using the principle of superposition to obtain the final solution for the response of the SSI system under the 1-g quasi-static excitation.

##### **9B.9.2.1 Analysis Cases**

Two sets of vertical 1-g static SSI analyses are performed for the unfactored and factored gravity inertia load conditions as discussed in Appendix 9B.A, Subsection 9B.6.3 and shown in Appendix 9B.A, Table 9B.6-1. Demands obtained from the 1-g static SSI analysis are from:

- The dead load from the self-weight and weight of permanent equipment and components
- The vertical fluid load on pool slabs from the weight of the water in the pools
- The static earth pressure load due to submerged weight of the soil and rock
- The surcharge lateral pressure load from the surrounding Power Block surface mounted foundations conservatively captures by assigning higher mass density properties as discussed in Appendix 9B.A, Subsection 9B.6.3.

In the model used to analyse the factored load combination, the submerged unit weight properties of the subgrade and surrounding structures is increased by 33% ( $1.6 / (1.2 = 1.33)$ ) to account for the higher factored lateral earth pressure loads.

### 9B.9.2.2 Frequencies of Analysis

The 1-g SSI analyses are performed for a total of seven frequencies of analysis from the first frequency of analysis with frequency equal to the frequency step,  $\Delta_f = 0.06$  Hz, to the cutoff frequency of 1.8 Hz. The low range of frequencies of analysis helps simulate a quasi-dynamic response of the SSI system under gravity load by avoiding response amplification at resonance frequencies.

### 9B.9.2.3 1-g Load Simulation

To simulate the 1-g gravity load, an acceleration time record in ACS SASSI is applied in the free-field location located at the ground surface in the form of a sine function at a very low frequency of 0.012 Hz. The low frequency content of the input motion helps avoid any dynamic effects in the calculated response due to resonance with the soil-structure system. Slightly different peak acceleration values of -1.005 g to 0.995 g are applied on the 1-g SSI analysis model in the negative and positive directions, respectively, to obtain amplitudes of the stress responses that correspond to the downward applied 1-g gravity load.

### 9B.9.2.4 Response and Demands from Dead Load and Earth Pressure Loads

This section reviews the unfactored demand results on the RB exterior wall below grade and SCCV wall because these are the primary resisting systems for both loads.

Figure 9B.9-1 illustrates the compression hoop forces acting on the RB exterior wall as a result of compressing static earth pressures. The change in distribution occurs at the floor elevations where the static earth pressure is transferred to the slabs. Figure 9B.9-2 illustrates the vertical compression forces in the RB exterior wall that increase as the depth increases reflecting the accumulation of the dead load within the RB integrated structures. Figure 9B.9-3 illustrates the horizontal bending moments about the tangential axis that peak at floor slab locations. Figure 9B.9-4 illustrates the radial out-of-plane shear on the RB exterior wall that peak at slab locations.

Figure 9B.9-5 illustrates that some of the static earth pressure is transferred from the RB exterior wall to the SCCV wall through the RB floor slabs as evident from the hoop force. The vertical compression force on the SCCV wall shown in Figure 9B.9-6 is more uniform than on the RB exterior wall (see Figure 9B.9-2). Horizontal moments and out-of-plane shear forces on the SCCV wall also peak at internal wall and at slab locations, respectively, similar to the RB exterior wall.

Trends observed with factored demands are similar to the trends observed for the unfactored demands presented in this section.

### 9B.9.3 Static and Quasi-Static Analysis

The analyses of static and quasi-static loads are performed on the FE models described in Appendix 9B.6, Subsection 9B.6.4. As stated in Subsection 9B.6.4, the analysis of the static and quasi-static loads that have global effect on the response of the integrated RB structural model are performed using the subgrade impedance sub-structuring method that accounts for the subgrade stiffness at the interfaces with the RB structure. Linear elastic contact springs connect the integrated RB structural model and subgrade FE models are developed using the subgrade stiffness impedance methodology. Stiffness properties are assigned to the contact springs to adequately represent the interaction mechanism between the RB structure, backfill, and subgrade.

Analyses performed to determine force/moment demands due to load cases for which the interaction with the surrounding soil is not important are performed on models with prescribed boundary conditions as described in Subsection 9B.6.4.

Table 9B.6-1 summarizes the modeling requirements for the various static and quasi-static analysis cases required for the design of the integrated RB structures.

### **9B.9.3.1 Analysis Cases**

Table 9B.9-7 presents the static and quasi-static load cases that are considered bounding for the design of the BWRX-300 integrated RB structures. As mentioned in Section 9.9, structural analyses evaluating the impacts of crane payload, tornado missiles, flooding, accidental loading outside of containment, and piping reaction loading, for which information is not available, will be performed as the design progresses.

The following subsections present responses and demands obtained from the load cases governing the design of the integrated RB structures.

### **9B.9.3.2 Response and Demands from Horizontal Pool Water Load Cases**

Horizontal pool water loads applied on the integrated RB model consist of the horizontal hydrostatic loads ( $F_h$ ) and sloshing and breathing mode hydrodynamic pressures due to seismic (DBE) excitation.

Pool water levels used for the hydrostatic load case listed in Subsection 9B.9.1.7 are considered to begin at the centreline of the refueling slab so that there is net zero hydrostatic pressure on interior pool walls that share a common wall with other pools. The hydrostatic pressure is applied as normal pressure with a gradient equal to the density of water at approximately 15 degrees Celsius ( $9.8 \text{ kN/m}^3$ ) multiplied by the distance below the centreline of the refueling deck slab. A hydrostatic pressure load is also applied to the containment closure head.

The deformed shape response of the pool walls due to hydrostatic loads is shown in Figure 9B.9-7. The outward expansion of the pool walls is consistent with the loading. The peak out-of-plane bending moments occur on the vertical and lower horizontal edges of the pool walls as expected for fixed-fixed-fixed-free two-way plates. The pool wall out-of-plane moments and associated transverse shear loads transfer to the walls and slabs that they intersect.

Hydrodynamic pressures on pool walls due to the vertical DBE excitation, referred to as breathing water pressure loads, are obtained by scaling the hydrostatic response with vertical response spectra.

### **9B.9.3.3 Responses and Demands from Containment Internal Pressure Load Cases**

The containment pressure loading listed in Table 9B.9-4 are applied normal to the SCCV wall, SCCV top slab, inner mat foundation, and containment closure head.

The integrated RB deformed shape response due to accidental pressure loading is shown in Figure 9B.9-8.

Results indicate a stress concentration on the pool wall as a result of the SCCV expanding upward. Higher bending moments occur in places where the expansion of the SCCV is constrained such as at the mat foundation and fuel pool walls and slabs.

### **9B.9.3.4 Responses and Demands from Groundwater and Rock Pressure Loads**

The lateral groundwater pressure is applied as an inward normal pressure with a gradient equal to the density of water at approximately 15 degrees Celsius ( $9.8 \text{ kN/m}^3$ ) multiplied by the distance starting from 3 m below grade down to the centreline of the common mat foundation.

The 31 meters of groundwater causing buoyancy uplift pressure on the mat foundation is applied as an upward normal load on the mat foundation model.

The in-situ residual rock pressure discussed in Subsection 9B.9.1.3 is applied as an inward normal pressure with a gradient applied circumferentially between 350 kPa and 250 kPa with linear interpolation in between. The rock pressure is applied to all nodes on the RB exterior wall at a depth of 23.9 m below the grade centreline and the centreline of the mat foundation.

Responses and demands due to groundwater and rock pressure loads are consistent with the loading.

The integrated RB deformed shape response due to groundwater lateral pressure is shown in Figure 9B.9-9. Groundwater pressure loads are transferred from the RB exterior wall to the SCCV wall through the RB floor slabs, which transfer the later loads in compression. Bending moment about the longitudinal axis of the RB outer peak at internal wall locations causing tension on the outer face of the RB exterior wall. Bending moments about the tangential axis of the RB exterior wall peak with outer face tension at the slab locations and inner face tension between slabs. Moments on the SCCV wall peak at RB wall and slab locations.

The uplift pressure causes minimal hoop forces in the RB exterior wall. The SCCV wall experiences compression forces as a result of buoyancy uplift. Positive bending on the mat foundation is observed resulting in tension on the top surface. The buoyancy uplift pressure loads are localized to the mat foundation and the walls at the floors. The magnitude of buoyancy uplift loads in walls is overcome with dead and live loads.

The integrated RB deformed shape response due to the additional rock lateral pressure is shown in Figure 9B.9-10. The force and moment demands for rock pressure are similar in distribution and order of magnitude to the groundwater lateral pressure demands.

#### **9B.9.4 Thermal Stress Analysis**

Thermal stress analyses are performed on the standalone FE models described in Appendix 9B.6, Subsection 9B.6.5. Modeling requirements for thermal load cases considered for the design of the integrated RB structures are presented in Table 9B.9-8.

##### **9B.9.4.1 Analysis Cases**

Table 9B.9-8 presents the thermal stress load cases considered for the boundary design of the BWRX-300 integrated RB structures. Excluded from this table is the accidental thermal loading outside of containment that is judged to be non-governing for bounding design of the structures.

Thermal loads are applied by assigning the temperature to the front and back faces of the element using structural temperature body loads. The thermal loads applied consider a stress-free reference temperature of 15°C. If the element is assigned different temperatures on either face, it results in a linear gradient.

Assumptions considered in the thermal stress load cases considered for the bounding design of the integrated RB structures include the following:

- Temperature loads are applied based directly on the specified bounding design temperatures neglecting the transient and steady-state effects of heat transfer which generally results in conservative estimates of thermal stress demands. This simplifying assumption will be verified by the results of the heat transfer analysis as the design progresses.
- The extreme winter conditions are considered to provide bounding thermal stress demands on the integrated RB structures since because of their larger thermal gradients when compared with summer ones. This assumption will be verified by subsequent analysis using precise thermal loading conditions.
- The DBA thermal stress evaluation considers only the condition when the containment experiences accidental temperature load inside the containment and the pool water temperature remains equal to the normal operating temperature. This assumption is expected to yield largest thermal stress responses on the SCCV wall and top slab. Additional DBA thermal stress analyses are to be performed as the design progresses considering differing temperature configurations associated with a variety of DBA conditions.

#### **9B.9.4.2 Response and Demands from Thermal Loads**

The deformed shapes for all normal operating and DBA thermal conditions during winter are shown in Figure 9B.9-11 and Figure 9B.9-12. The expansion and contraction of the outer RB is consistent with the effect of the differential temperatures exterior and interior to the RB. The RPV pedestal and bioshield appear to experience the highest expansion deformations because of the higher temperatures in the SCCV.

The accidental temperature load case during winter is used as representative of the other thermal loads since it results in the highest overall demands on the structures. In-plane forces and out-of-plane bending moments due to the accidental temperature load case during winter are shown in Figure 9B.9-13 and Figure 9B.9-14, respectively.

Figure 9B.9-13(a) illustrates the hoop forces in the concentric walls that peak in tension at the refueling floor level because of the contraction of the exterior walls and expansion of pool walls. Figure 9B.9-13(b) illustrates the vertical forces in the concentric walls and tangential in-plane forces in the slabs. The SCCV wall is generally in compression because of the constraints from the slabs and pool walls above. The inner mat foundation is also in high compression. The out-of-plane moments peak at the ICS pool wall behind the fuel pool (Figure 9B.9-14(a)) and at the RB exterior wall at grade (Figure 9B.9-14(b)). The SCCV is under biaxial bending throughout its height as a result of constraints from the RB slabs. The accidental thermal demands during winter are found to be about two times higher than the normal operating winter demands.

#### **9B.9.5 Conclusions**

Selecting bounding conditions for most design parameters provides conservative bounding demands for the design of the BWRX-300 Seismic Category A structures at the DNNP site. Stress concentrations observed in some of the members force and moment results are due to singularities in the FE solution and may be interpreted as unrealistically high demands that, in turn, lead to the observance of high demand-to-capacity ratios in the design evaluations.

Of all the load cases considered in the bounding design of the integrated RB structures, the structural demands obtained from the accidental thermal load case exceed the average peak demands of the next highest tier of loads by a minimum factor of two. The second highest tier of structural demands includes normal operating temperature demands, testing pressure demands, and accidental pressure demands. The third tier of demands with average peaks at approximately half of the second-tier demands are obtained from the unfactored dead and static earth pressure loads, hydrostatic groundwater loads, rock pressure loads, and accidental seismic torsion loads. The lowest tier includes average peak structural demands that are equal to or under one-quarter of the Tier 2 demands. This fourth tier is obtained from live loads, buoyancy load, normal operating pressure loads, sloshing loads, design snow loads, extreme snow loads, wind and tornado wind loads.

Peak membrane forces and out-of-plane bending moments in the SCCV (inner) mat foundation are dominated by the accidental temperature demands. Accidental temperature demands are approximately double peak demands from the normal operating temperature load case with the latter being approximately four times higher than the next largest demand from rock pressure and groundwater demands. All other mechanical demands are significantly less than the groundwater demands. Out-of-plane shear force demands in the SCCV mat foundation are dominated by the accidental and testing pressures followed by the 1-g static and accidental temperature load cases.

The accidental temperature load case also causes the largest membrane forces in the SCCV top slab, followed by the normal temperature load case, followed by the accidental and testing pressure load cases. All other mechanical demands are marginally less than the pressure demands with the largest coming from the static 1-g load case. Out-of-plane bending moments in the SCCV top slab are dominated by the accidental temperature demands which are approximately double the demands from

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

the accidental pressure. The normal operating temperature load case provides the next largest demands, with minimal demands obtained from all other mechanical loads. Out-of-plane shear demands in the SCCV top slab occur in localized areas due to the accidental temperature load case and to a lesser extent, the accidental pressure, or testing pressure load cases; followed by demands from the 1-g static dead and static earth pressure loads.

The SCCV wall membrane forces are dominated by accidental temperature demands which are approximately double the demands from normal operating temperature, accidental, or testing pressure. The next largest demands in the SCCV wall come from the 1-g static load case followed by groundwater, or rock pressure. All other mechanical demands are significantly less than these demands. Out-of-plane bending moments and out-of-plane shear forces in the SCCV wall are dominated by the accidental temperature load case followed by the normal operating temperature load case at approximately half. The accidental pressure and testing pressure, groundwater, rock pressure, and 1-g static demands are all within range of one another.

The RB (outer) mat foundation is dominated by the demands from the accidental temperature load case followed by the normal operating load case at approximately half the peak accidental temperature demands. The rock pressure and groundwater demands follow at approximately 1/5th of the accidental temperature load case, followed by the 1-g static and accidental pressure load cases. The out-of-plane bending moments and out-of-plane shear forces in the outer mat foundation are dominated by the accidental temperature load case followed by the normal operating temperature load case at approximately half. The rock pressure, groundwater, and 1-g static load case demands are each approximately half or less than the normal operating temperature demands.

Peak membrane demands in the RB exterior wall from the accidental temperature, rock pressure, groundwater, and 1-g static load cases are all in the same range, with the next largest demands coming from the accidental pressure and incidental torsion at about a quarter of the primary demands. Out-of-plane moment demands in the RB exterior wall from the accidental temperature, rock pressure, groundwater, and 1-g static load cases are also all in the same range, with the next largest demands coming from the normal operating temperature and incidental torsion. Out-of-plane shear demands in the RB exterior wall are dominated by the 1-g static and incidental torsion load cases, with the next largest demands coming from the accidental thermal, rock pressure, and groundwater pressure load cases.



**Table 9B.9-1: Equivalent Linear Static Subgrade Profiles**

Analysis Case	Bounding Parameters Used
1-g SSI analysis	UB $\gamma$ , LB $E_{st}$ and UB $\nu_{st}$
Thermal analysis	UB $E_{st}$ and UB $\nu_{st}$
Mechanical load case analysis	LB $E_{st}$ and LB $\nu_{st}$

- (1)  $\gamma$  = Effective unit weight for subgrade materials below groundwater table  
(2)  $E_{st}$  = Young Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions  
(3)  $\nu_{st}$  = Poisson's ratio representative of at-rest lateral earth pressure conditions

**Table 9B.9-2: Bounding Tornado Design Parameters**

Tornado Characteristics	Reactor Building Structure
Maximum wind speed, $V_{max}$	371 km/hr
Maximum rotational speed, $V_{Rm}$	295 km/hr
Translational speed, $V_{tr}$	76 km/hr
Radius, $R_m$	45.7 m
Pressure drop, $P_a$	8.3 kPa
Rate of pressure drop, $p_t$	3.7 kPa/s
Acceptance Limit State	Linear Elastic

**Table 9B.9-3: Tornado Missile Spectrum and Maximum Horizontal Speed**

Missile Type	Dimensions	Mass	Horizontal Velocity	Vertical Velocity
Schedule 40 Pipe	0.168 m diameter x 4.58 m long	130 kg	147.6 km/h	99 km/h
Automobile	5 m x 2 m x 1.3 m	1810 kg	147.6 km/h	99 km/h
Solid Steel Sphere	25.4 mm diameter	0.0669 kg	28.8 km/h	19.3 km/h

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B.9-4: Containment Pressure Loading**

Pressure Load		Gauge Pressure kPa(g)
Normal positive pressure	$P_{o,+}$	18.7
Normal negative pressure	$P_{o,-}$	-14
LOCA Accident Peak Pressure	$P_a$	413.7
Test Peak Pressure	$P_t$	475.73

**Table 9B.9-5: Summer/Winter Thermal Conditions in Reactor Building and SCCV During Normal Operations**

REGION		SUMMER (or Maximum)	WINTER (or Minimum)
Outside Containment (RB)	Main Steam/Feedwater Penetration Room	40°C	40°C
	Interior	40°C	10°C
	RB Outdoor Temperature	40°C	-40°C
	Pools	43°C	43°C
	Ground	15.5°C	15.5°C
Inside containment (SCCV)		65°C	

**Table 9B.9-6: Summer/Winter Thermal Conditions in Reactor Building and SCCV During DBA**

	REGION	SUMMER (or Maximum)	WINTER (or Minimum)
Outside Containment (RB)	Main Steam/Feedwater Penetration Room	40°C	40°C
	Interior	40°C	10°C
	RB Outdoor Temperature	40°C	-40°C
	Pools*	100°C	100°C
	Ground	15.5°C	15.5°C
Inside containment (SCCV)		165.5°C	

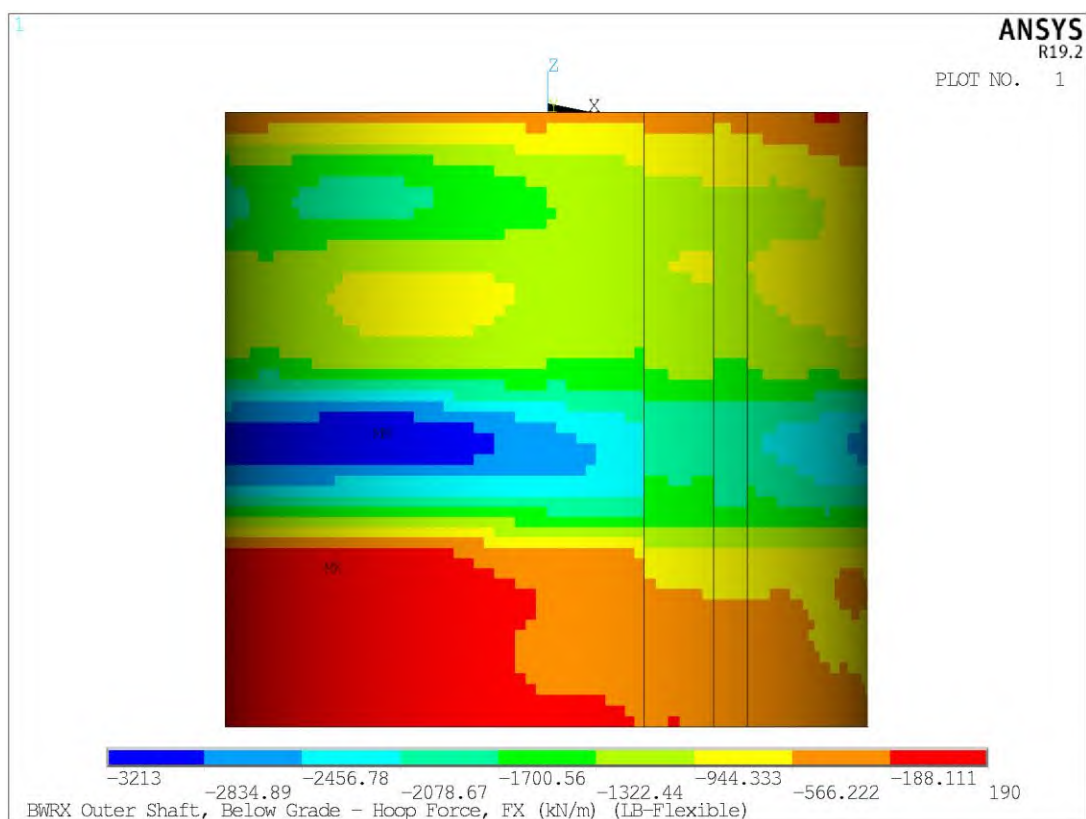
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

**Table 9B.9-7: Significant Static and Quasi-Static Load Cases**

Loading Type	Load Case	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Live load on floors	L	Effective BE	LB Static massless LB Poisson's Ratio LB Young's Modulus	Rigid in normal directions Soft in the other 2 directions	Rigid in all three directions	Standalone RB model
Hydrostatic and seismic hydrodynamic breathing mode pressures on pool walls	F <sub>h</sub> or E <sub>b</sub>					
Design wind load on above grade exterior wall and roof	W					
Tornado wind load on above grade exterior wall and roof	W <sub>t</sub>					
Normal operating and SIT uniform containment pressure	P <sub>0</sub> , P <sub>t</sub>					
Accidental uniform containment pressure	P <sub>a</sub>	LB for Containment and exterior pool walls and Effective BE elsewhere		Soft in all three directions	Fixed mat foundation nodes	
Accidental seismic torsion	E <sub>t</sub>	Effective BE			Rigid in all three directions	
Seismic hydrodynamic sloshing load on pool walls	E <sub>s</sub>					
Normal snow on RB roof	S					
Extreme precipitation (snow) pressure load on RB roof	S <sub>x</sub>					
Groundwater hydrostatic pressure on mat foundation (H <sub>b</sub> ) and RB exterior wall (H <sub>g</sub> ) below groundwater level	H <sub>g</sub> , H <sub>b</sub>					
Additional lateral rock pressure on RB exterior wall	H <sub>r</sub>					

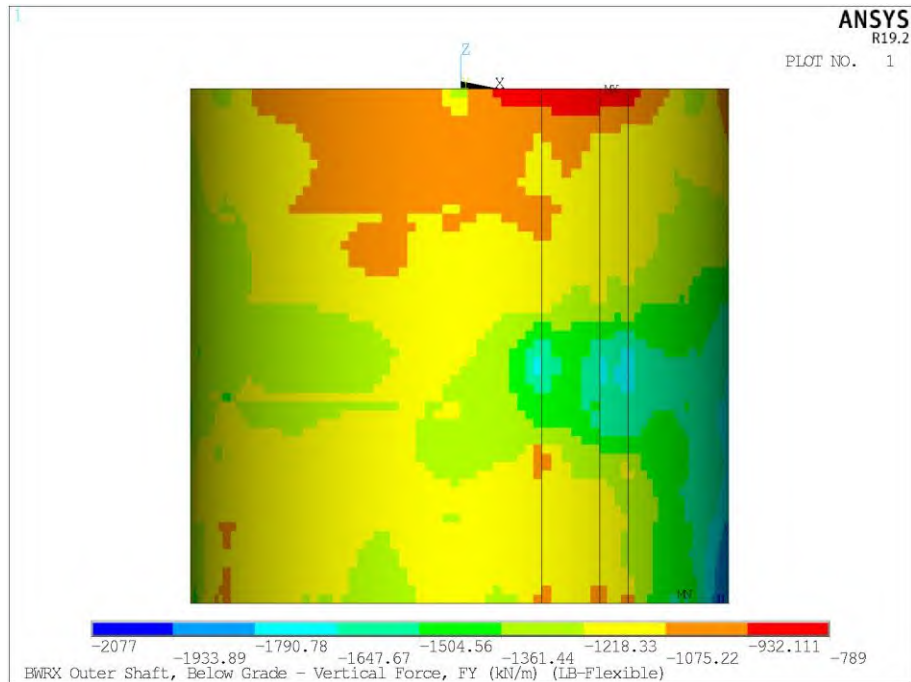
**Table 9B.9-8: Significant Thermal Load Cases**

Loading Type	Load Case	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Normal Operating thermal loads	T <sub>0</sub>	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	UB static massless UB Poisson's Ratio UB Young's Modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
Accidental thermal loads on Containment	T <sub>a</sub>		LB for cracked SC elements and effective BE elsewhere				

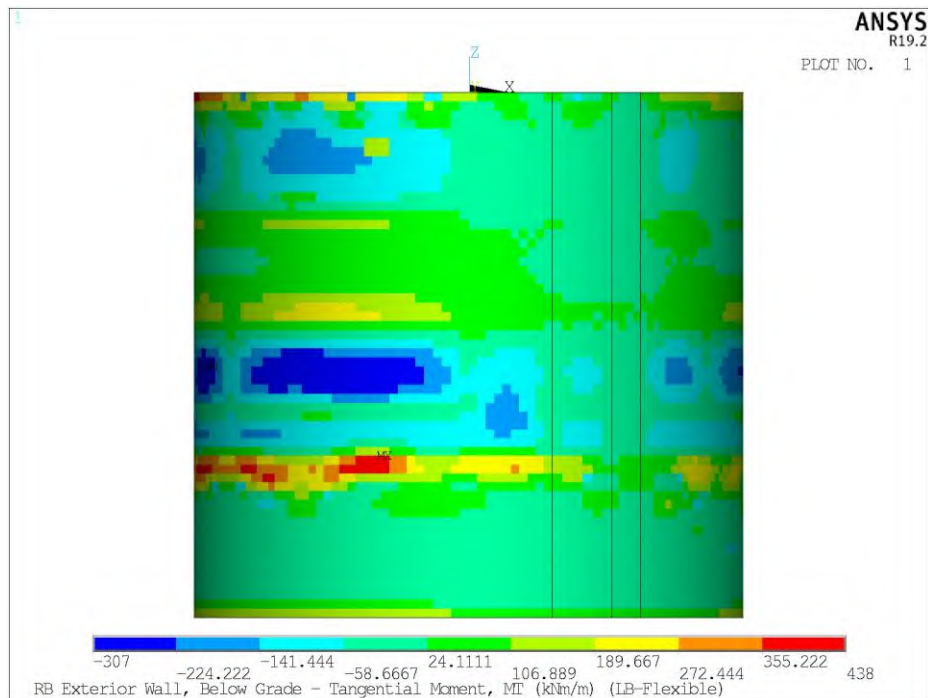


**Figure 9B.9-1: Compression Hoop Forces on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads**

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION

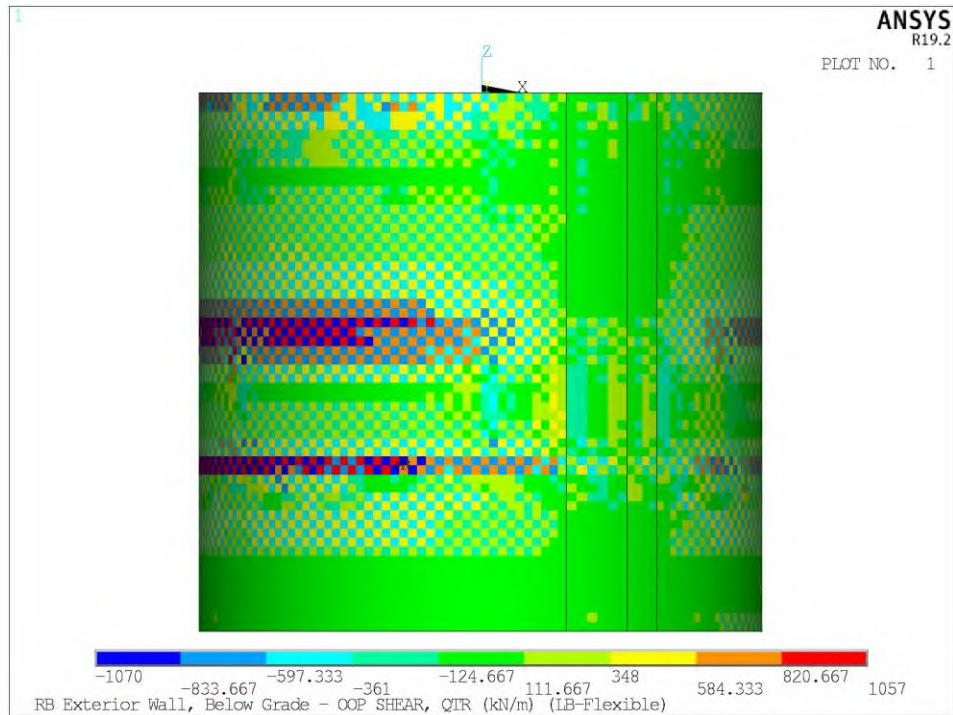


**Figure 9B.9-2: Vertical Compression Forces on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads**

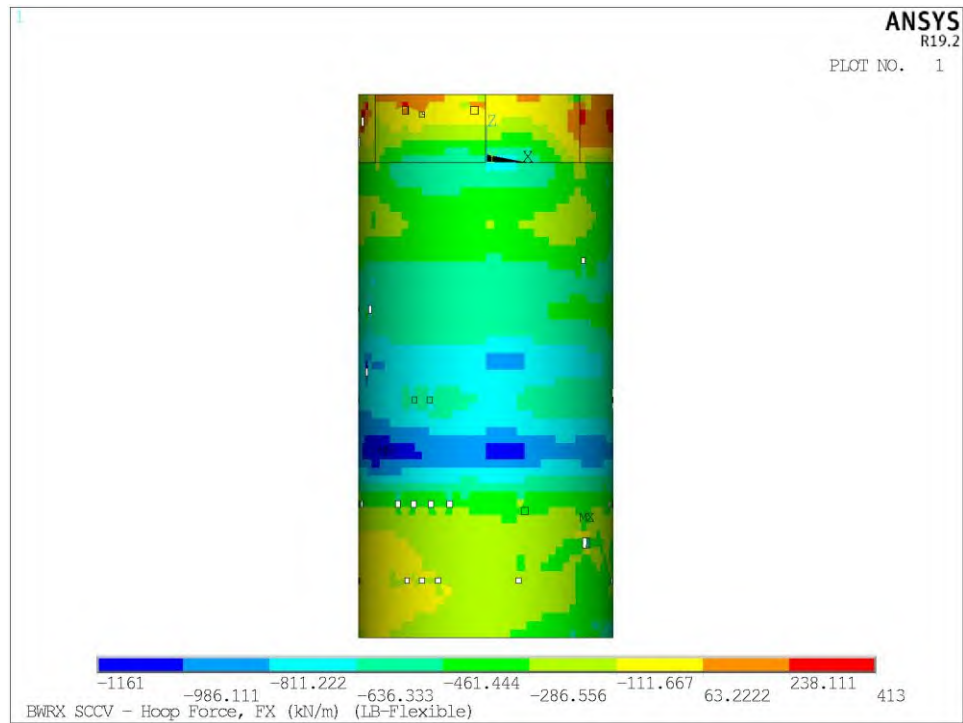


**Figure 9B.9-3: Horizontal Bending Moments on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads**

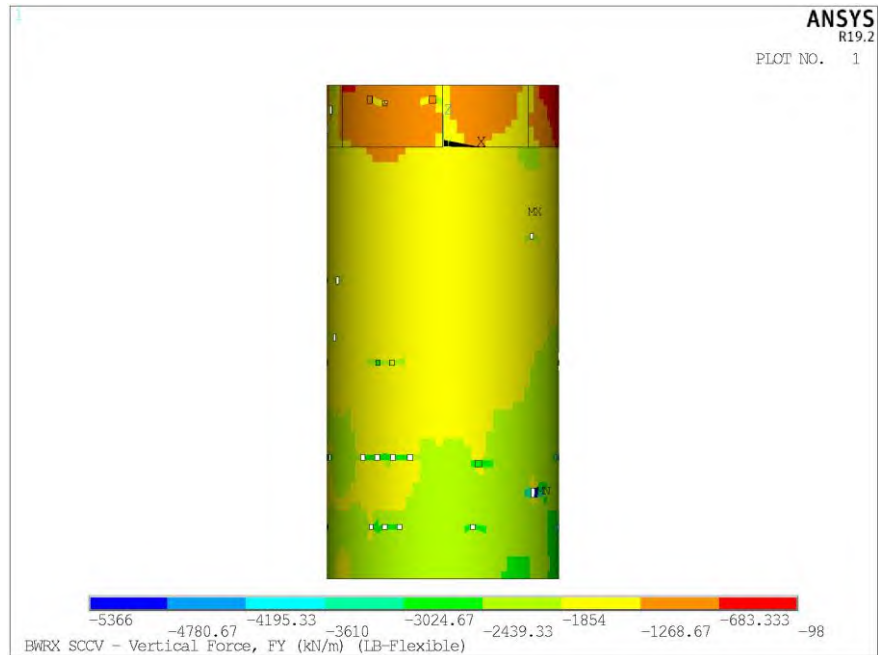
NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



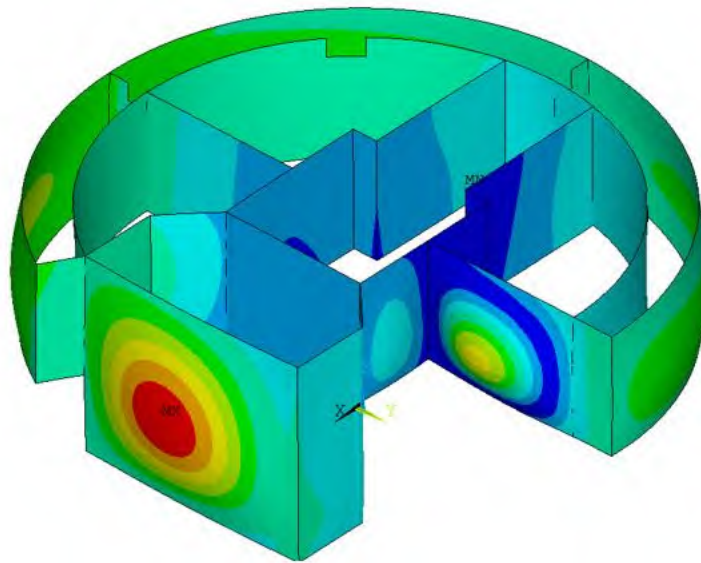
**Figure 9B.9-4: Out-of-Plane Shear on the Reactor Building Exterior Wall from Unfactored Dead and Static Earth Pressure Loads**



**Figure 9B.9-5: Compression Hoop Forces on the SCCV Wall from Unfactored Dead and Static Earth Pressure Loads**

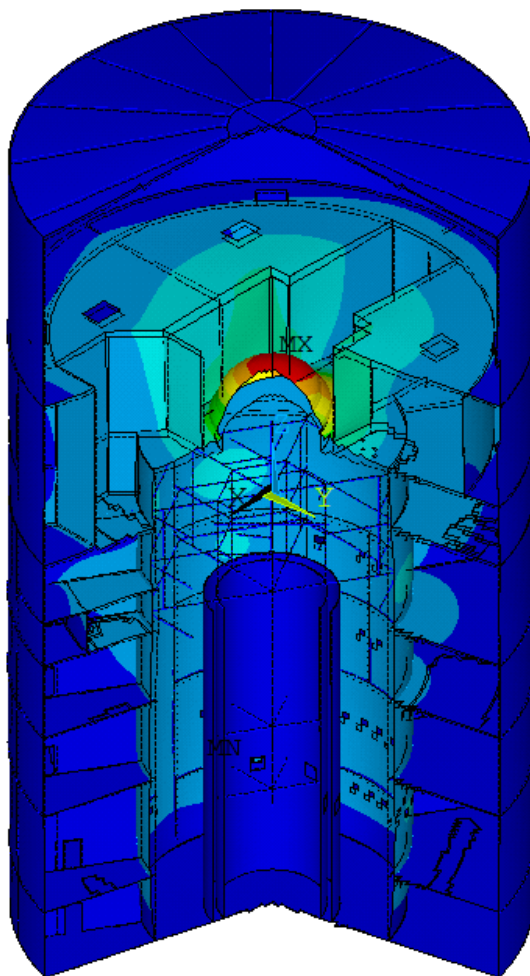


**Figure 9B.9-6: Vertical Compression Forces on the SCCV Wall from Unfactored Dead and Static Earth Pressure Loads**



**Figure 9B.9-7: Pool Walls Deformed Shape due to Hydrostatic Loads**

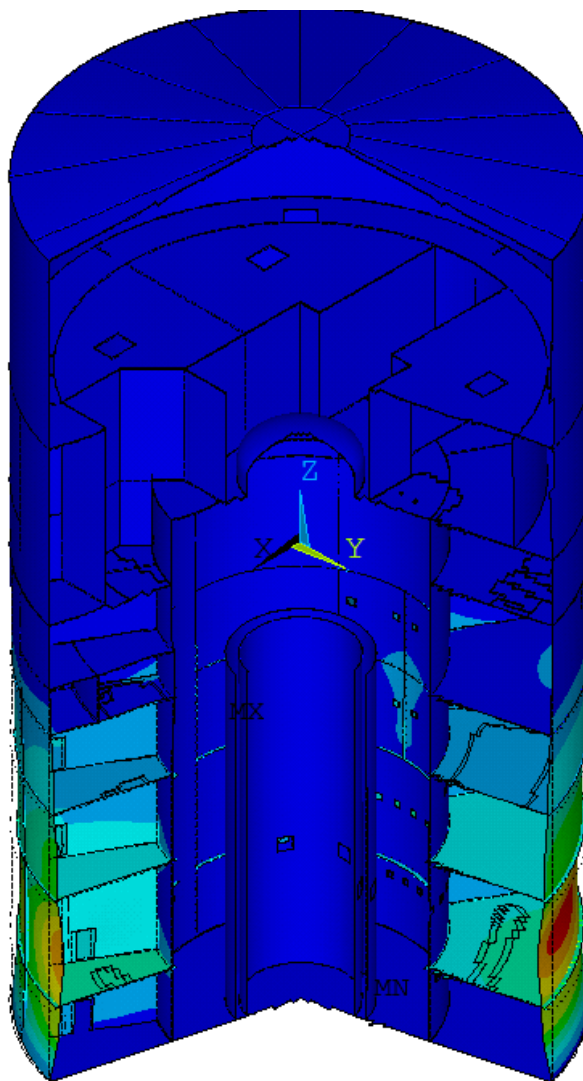
\*For illustration purposes only



**Figure 9B.9-8: Deformed Shape of Integrated Reactor Building due to Accidental Pressure**

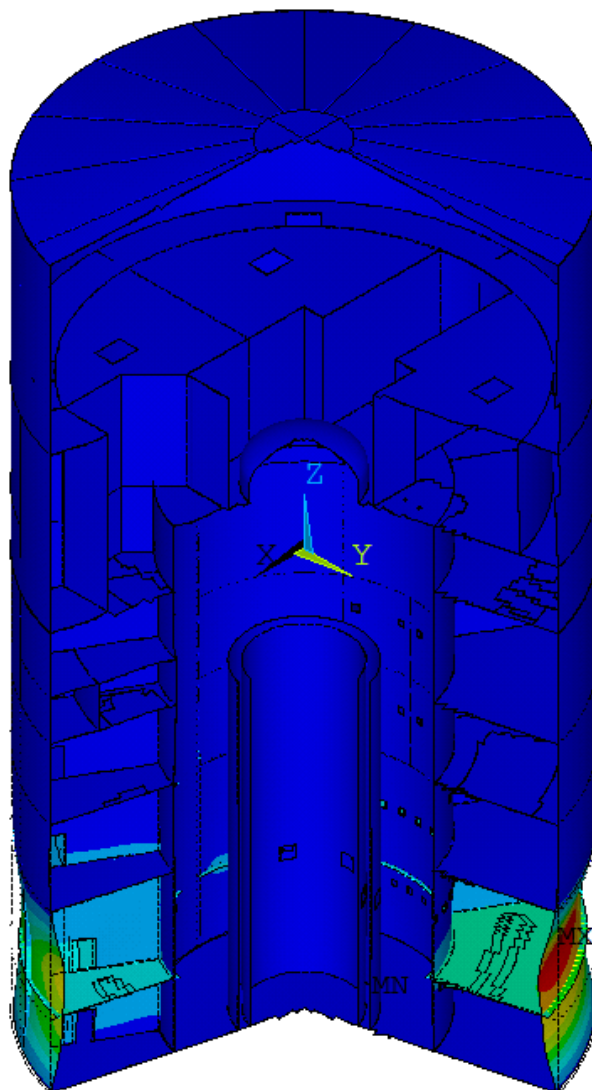
\*For illustration purposes only





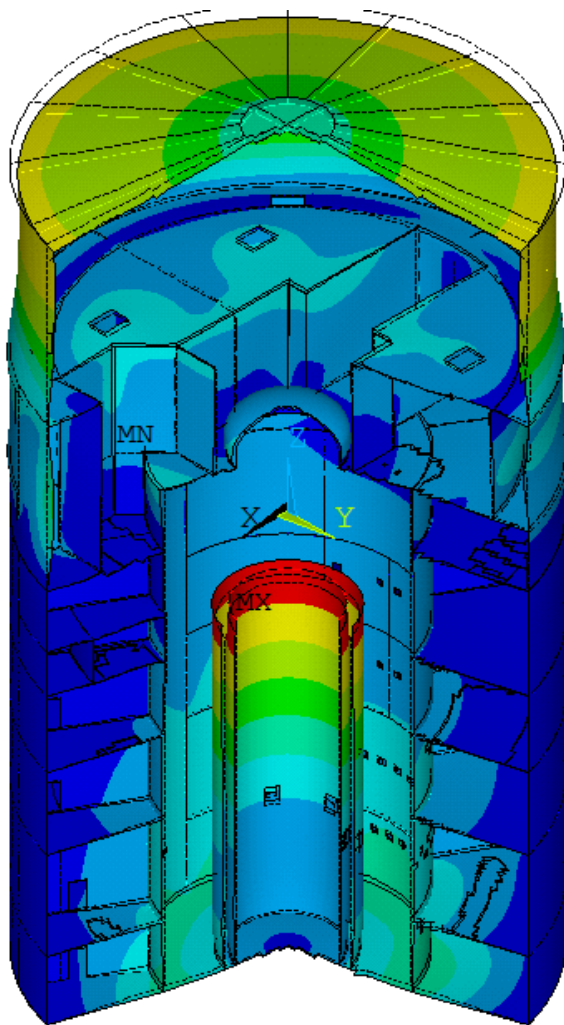
**Figure 9B.9-9: Deformed Shape due to Groundwater Pressure on Reactor Building Exterior Wall**

\*For illustration purposes only



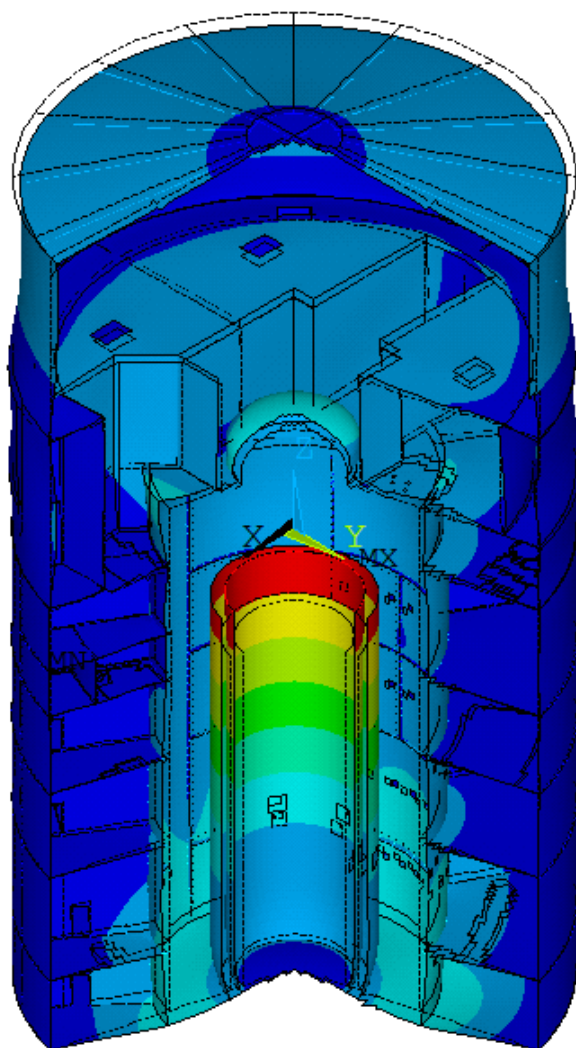
**Figure 9B.9-10: Deformed Shape due to Additional Rock Pressure on Reactor Building Exterior Wall**

\*For illustration purposes only



**Figure 9B.9-11: Deformed Shape of Integrated Reactor Building Under Normal Operating Winter Temperatures**

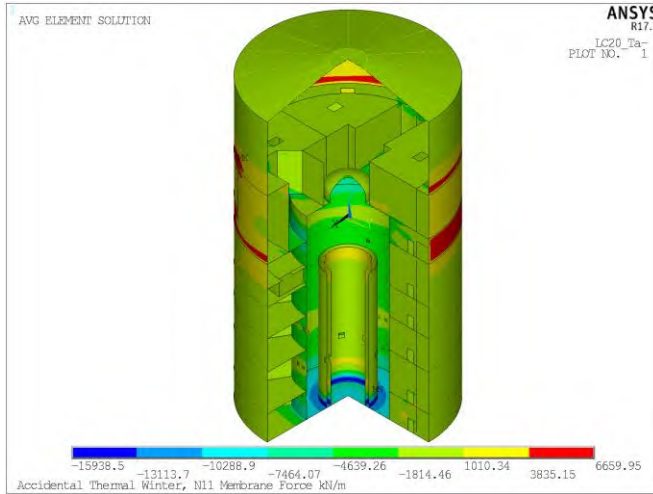
\*For illustration purposes only



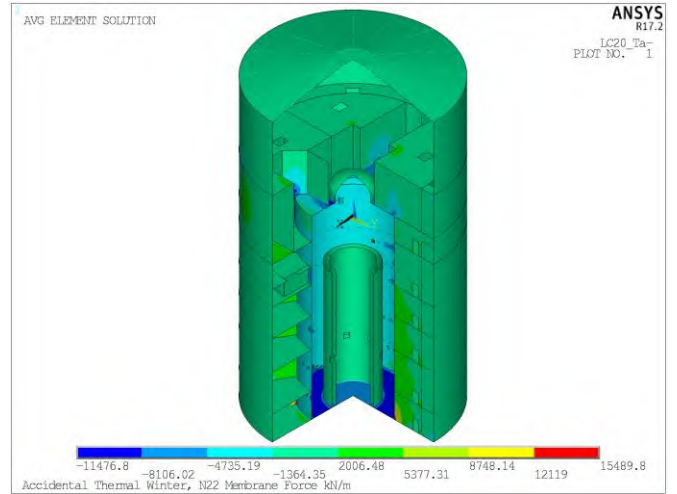
**Figure 9B.9-12: Deformed Shape of Integrated Reactor Building Under DBA temperatures during Winter**

\*For illustration purposes only

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



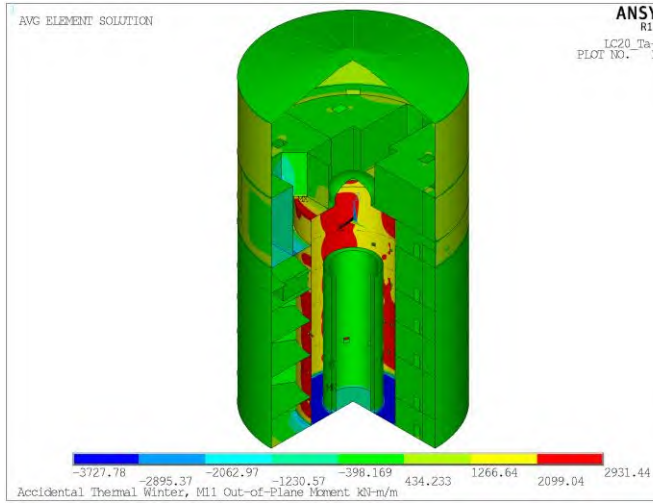
(a)



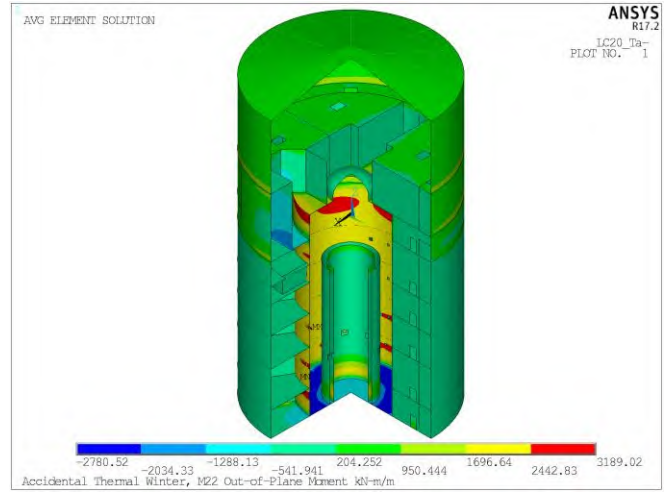
(b)

**Figure 9B.9-13: In-Plane Forces due to DBA Temperatures During Winter**

NEDO-33959 REVISION 2  
NON-PROPRIETARY INFORMATION



(a)



(b)

**Figure 9B.9-14: Bending Moments due to DBA Temperatures During Winter**

## **APPENDIX 9B.E – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT**

### **9B.10 Introduction**

This appendix focuses on the design evaluation of the SCCV portion of the containment as discussed in Subsection 9B.2.1.10.2 and provides bounding estimates of the available margins for the structure.

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendices 9B.B and 9B.D, performed using the one-step approach and the FE models discussed in Appendix 9B.A. The design demands are obtained from bounding design load combinations selected to assess the operational and accidental structural demands on the structure.

Demand-to-capacity ratios are calculated in accordance with NEDC-33926P.

#### **9B.10.1 SCCV Design Methodology**

The design, detailing and geometric requirements for the SCCV are in accordance with Section 4.0 of NEDC-33926P.

Table 9B.10-1 shows the design parameters used in the Steel Bricks™ design evaluation.

#### **9B.10.2 Bounding Load Combinations and Design Demands**

The evaluation of the SCCV is based on results from four bounding load combinations, SCCV-1, SCCV-3, SCCV-7 and SCCV-11, presented in Table 9B-1. SCCV-1 is for testing category at service condition, SCCV-3 for normal category at service condition, SCCV-7 for abnormal load category at factored condition and SCCV-11 for abnormal/extreme environmental load category at factored condition.

Per NEDC-33926P, In-plane membrane forces, out-of-plane moments, and out-of-plane shear forces may be obtained by averaging the results over panel sections that are no larger than twice the section thickness in length and width. Near openings and penetrations, and in connection regions, the required strength may be averaged over panel sections no larger than the section thickness in length and width.

For evaluation purposes, design demands are conservatively presented on element-by-element basis without averaging of the results across adjacent elements as allowed by the NEDC-33926P.

#### **9B.10.3 Structural Capacity of SCCV Steel Bricks™**

Steel Bricks™ sections capacities are computed, and the design evaluated per NEDC-33926P, Section 4.0. Acceptance criteria for the design of the SCCV are per Chapter 3, Table 3.5-4.

Individual strength checks include axial (tension and compression), flexural and shear (in-plane and out-of-plane) strength checks. Combined design checks include out-of-plane shear interaction and in-plane forces and out-of-plane moments interactions.

Force and moments induced in the SCCV Steel Bricks™ by factored and service load combinations are calculated in the sections radial and meridional directions in unit widths. Demands are then divided between the Steel Bricks™ two notional halves, each consisting of single faceplate and half concrete thickness, as discussed in NEDC-33926P. To evaluate the limitations of the Steel Bricks™ steel plates, principal stresses in steel plates are used to calculate the von Mises stress for each of the plates to ensure that this stress does not exceed the allowable stresses for factored and service load combinations summarized in Chapter 3, Table 3.5-4. The principal stresses in the concrete infill are also checked to ensure they too do not exceed the allowable stresses listed in Chapter 3, Table 3.5-4.

The principal stresses for concrete infill and von Mises stresses for steel plates are used to evaluate the in-plane membrane and out-of-plane moment interactions.

#### **9B.10.4 Structural Design Evaluation**

The evaluation of the SCCV structure is performed by calculating and plotting demand-to-capacity ratios for the individual loads and interaction checks discussed in Subsection 9B.10.3 for the SCCV wall, top slab and inner mat foundation. The demand-to-capacity ratios identify critical locations and vulnerabilities for specific limit states. For each load combination, multiple patterns of load application are considered, depending on their severity for a particular limit state.

The evaluation is performed for factored, or service loads classified as “Primary” for (1) membrane forces alone and (2) membrane forces plus bending moments against the allowable for factored or service primary forces class. The evaluation is performed once more whenever volume changing secondary forces (e.g., Thermal loading) are in the load combinations for the allowable under factored or service primary plus secondary forces class. For factored loads, where volume changing secondary forces are included, the strain limit for mechanical (net) strain is twice the yielding strain. However, the total strain (mechanical plus volume changing strain) is compared to the same limit as a conservative approach. The highest demand-to-design ratio for the steel plates and concrete infill, if any, are reported in this appendix.

Per NEDC-33926P, Section 4.0, SCC-3523.1, interaction checks are not performed if any of the two shear components on each direction of an element is less than the concrete capacity. In this case a zero demand-to-capacity ratio is reported in the plots.

Figure 9B.10-1 through Figure 9B.10-7 present bounding demand-to-capacity ratios for the SCCV wall, top slab and inner mat foundation for the selected load combinations listed in Subsection 9B.10.2.

The demand-to-capacity ratio plots developed for the testing load category (SCCV-1) indicate that the design of the SCCV structure includes adequate safety margins as demonstrated by Figure 9B.10-1 and Figure 9B.10-2. Figure 9B.10-1 and Figure 9B.10-2 present the in-plane forces and out-of-plane moment interaction checks for the steel plates of the SCCV wall and top slab, respectively. As shown in Figure 9B.10-1, the maximum demand-to-capacity ratio recorded for the in-plane forces and out-of-plane moment interaction for the steel plates of the SCCV wall is 0.7, except at vicinity of openings due to singularities in the FE solution. From Figure 9B.10-2, the maximum demand-to-capacity ratio recorded for the in-plane forces and out-of-plane moment interaction for the top slab’s steel plates is 0.6. For both structural components and for the inner mat foundation, the demand on the concrete infill is minimal.

Figure 9B.10-3 presents the highest demand-to-capacity ratio recorded in the interaction checks of key SCCV components for the normal load category. The demand-to-capacity ratio of 0.6 is computed for the concrete infill of the inner mat foundation.

Figure 9B.10-4(a) indicates that the out-of-plane flexural demand in the inner mat foundation under the abnormal load category is generally small, with a demand-to-capacity ratio less than 0.8 observed at the intersection with the RPV pedestal and bioshield. The in-plane forces and out-of-plane moment interaction checks for the mat’s concrete infill shown in Figure 9B.10-4(b) also record a maximum demand-to-capacity ratio of 0.6. Demands on the mat’s steel plates are minimal with demand-to-capacity ratios governed by the primary component allowable limit. Figure 9B.10-5 and Figure 9B.10-6 present the out-of-plane shear checks that govern the design of the SCCV wall and top slab. As demonstrated by these figures, the demand-to-capacity ratios for the out-of-plane shear in these structures are generally less than 0.6, with some regions indicating a no requirement for biaxial shear interaction check. The small stress concentrations observed in Figure 9B.10-5 and Figure 9B.10-6 occur within the intersecting depth of the mat foundation (in Figure 9B.10-5) and at the slab intersection with the different pool walls and containment head (in Figure 9B.10-6) and do not affect the design.



Figure 9B.10-7 presents the out-of-plane shear check at the SCCV top slab for the abnormal/extreme environmental load category. As shown in this figure, the maximum demand-to-capacity for out-of-plane shear in the structure is generally less than 0.5, except for the stress concentrations at the intersection with the different pool walls and containment head due mainly to singularities in the FE solution.

The evaluation results presented in this appendix show that the overall demand-to-capacity ratio for the SCCV structure is less than one. This demonstrates that the design of the SCCV structure is robust and includes adequate safety margins in conformance with REGDOC-2.5.2, Section 7.15.1.

#### **9B.10.5 Critical SCCV Steel Brick™ Connections**

The SCCV wall is connected to the RB wing walls and floor slabs at various levels and shares the mat foundation with the RB.

SCCV wall-to-mat foundation and SCCV top slab-to-wall full-strength connections are designed for a demand of 125% of the smaller of the corresponding nominal strengths of the connected parts. SCCV wall-to-wing walls and SCCV wall-to-floors connections are identical to those connecting the floors to the RB exterior wall discussed in Appendix 9B.G, Subsection 9B.12.5.4.

##### **9B.10.5.1 SCCV Wall-to-Mat Foundation Connection**

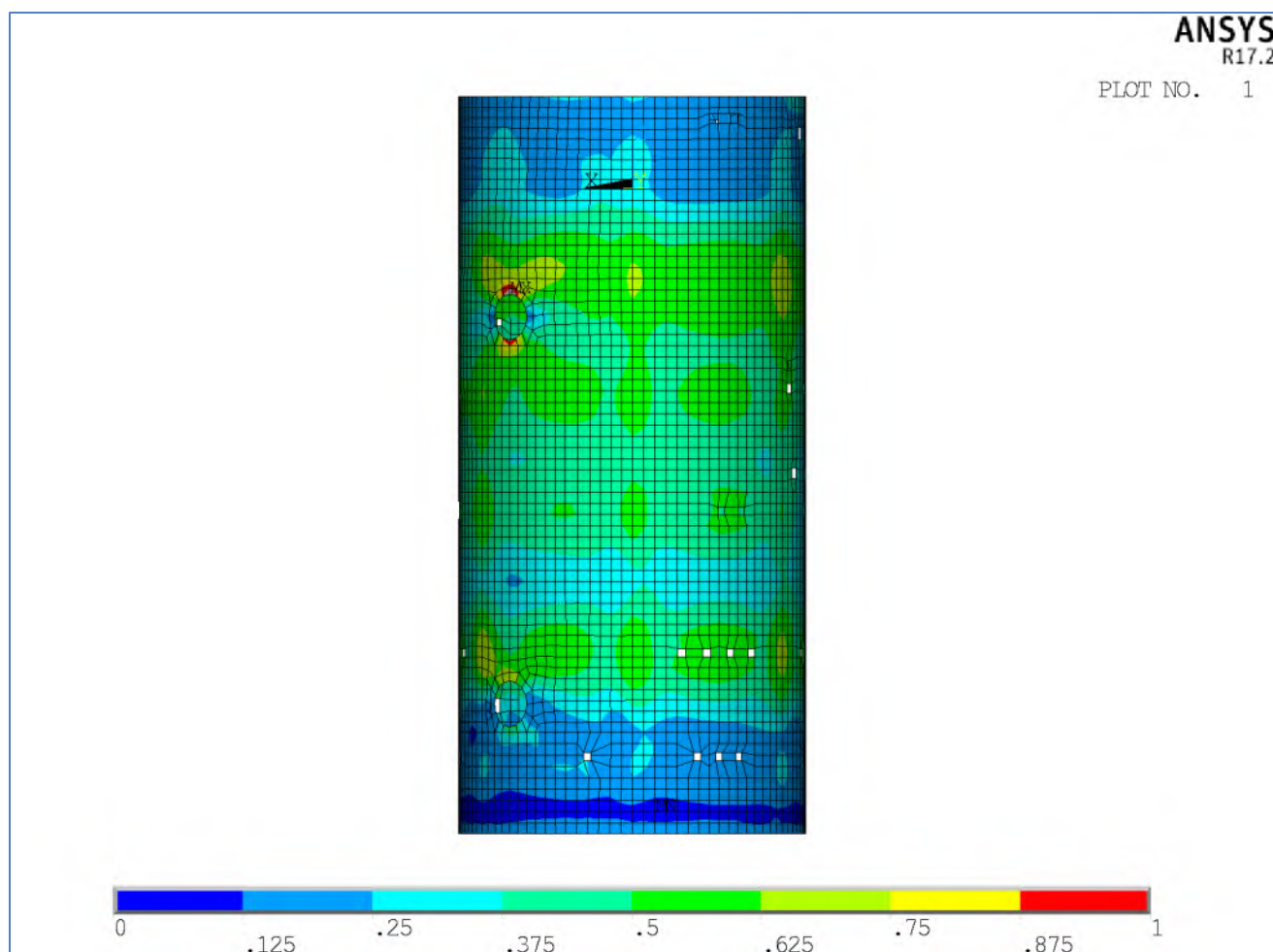
The SCCV wall-to-mat foundation connection, referred to as T connection, is subject to LOCA accident conditions (e.g., accidental pressure ( $P_a$ ) and accidental temperature ( $T_a$ )).

The design of the Steel Bricks™ T connection is performed using the full-strength design philosophy. In this approach, the connection is considered to sustain loads until the full-strength of the weaker of the two connected parts is achieved. The T connection is idealized using the model shown in Figure 9B.10-8. In this model, the two ends of the SCCV wall at the T connection are considered fixed. Design demands and required strength for each individual demand acting on the connection are calculated by identifying the applicable force transfer mechanism and evaluating the potential failure mode or mechanism of the connection. Connection capacities are taken as 1.25 times the smaller of the two connected components capacities.

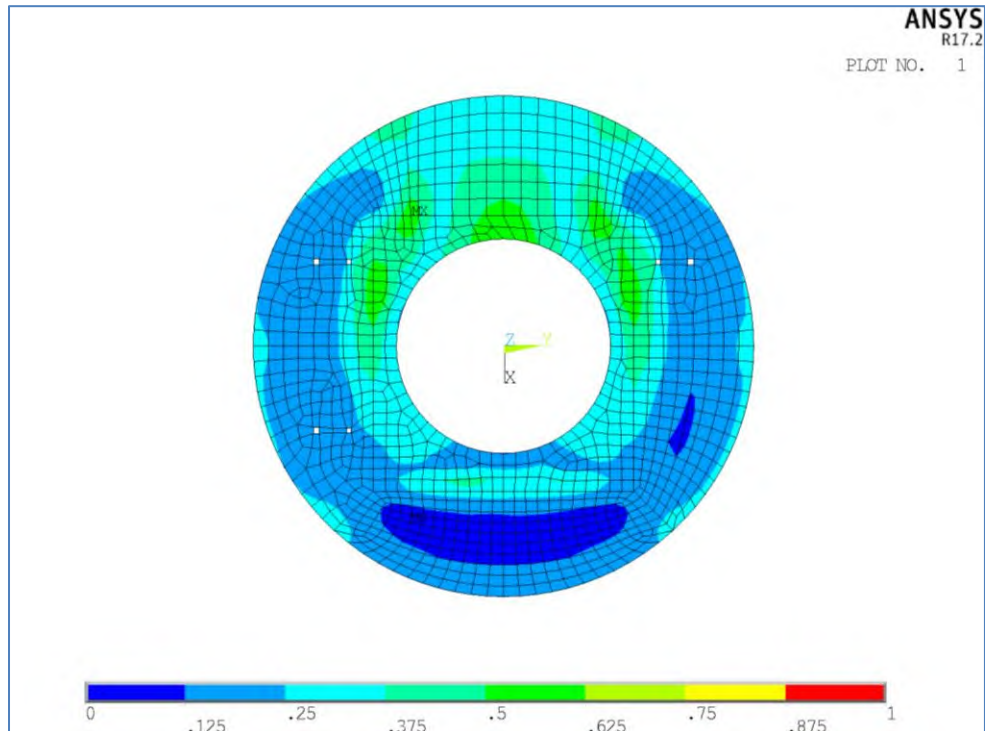
**Table 9B.10-1: Design Parameters for the SCCV Steel Bricks™**

SCCV Structural Component	Member Thickness mm	Faceplate Thickness, mm	Steel Faceplate Grade	Yield Strength (MPa)	Tensile Strength (MPa)	Diaphragm Spacing (Brick width) (mm)
SCCV wall	914	15.0	ASME SA738 Grade B	414 <sup>1</sup>	448	457
Inner mat foundation	1219	12				609
SCCV top slab	914	15				457

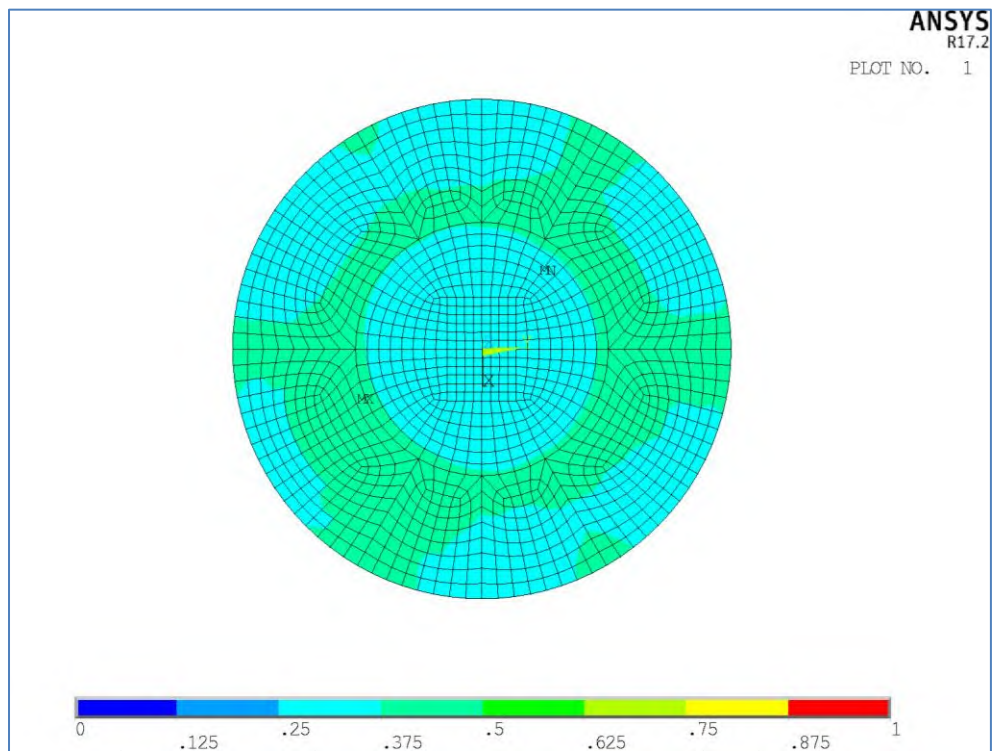
(1) For Load combinations with accidental thermal loading, a reduced yield strength of 353 MPa is used for the ASME SA738 Grade B steel



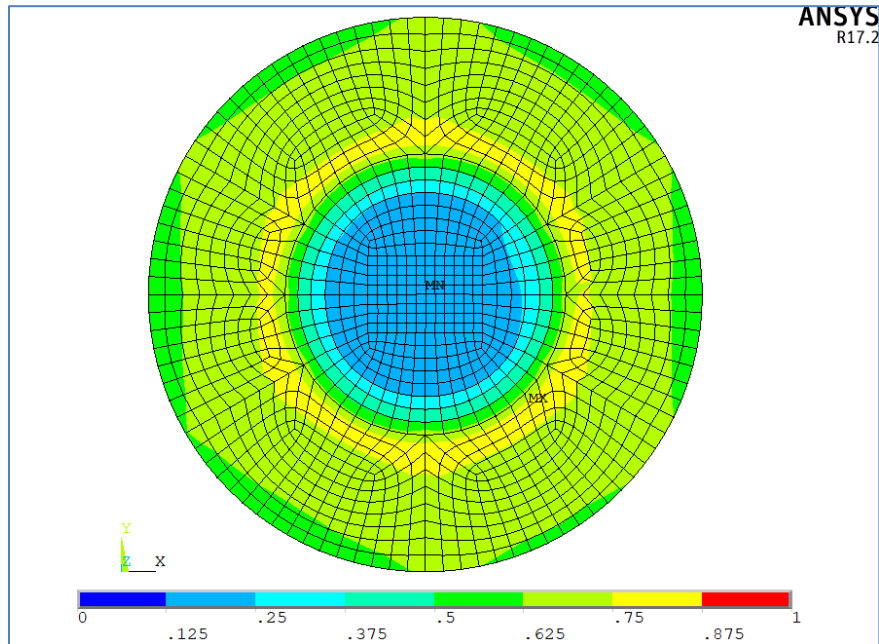
**Figure 9B.10-1: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for SCCV Steel Plates Under Testing Load Category (SCCV-1)**



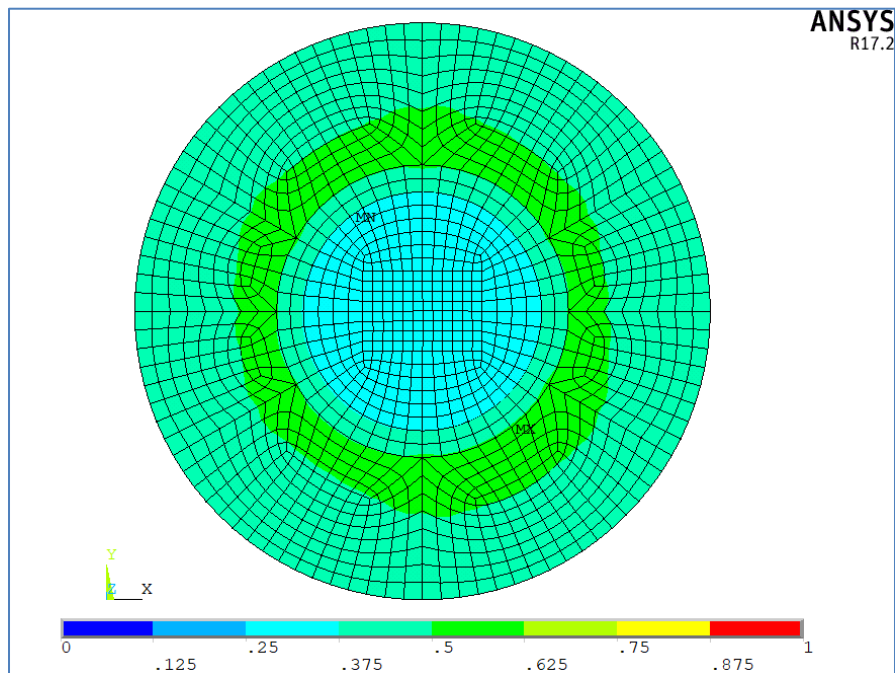
**Figure 9B.10-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the SCCV Top Slab Steel Plates Under Testing Load Category (SCCV-1)**



**Figure 9B.10-3: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the Inner Mat Foundation Concrete Infill Under Normal Load Category (SCCV-3)**

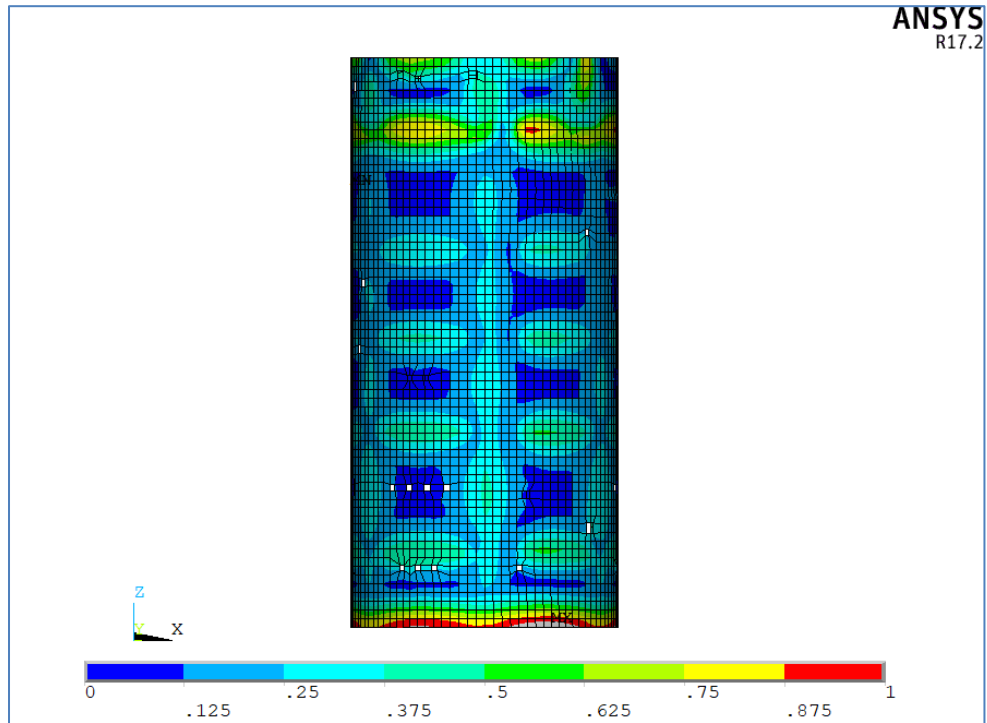


(a) Out-of-Plane Flexure

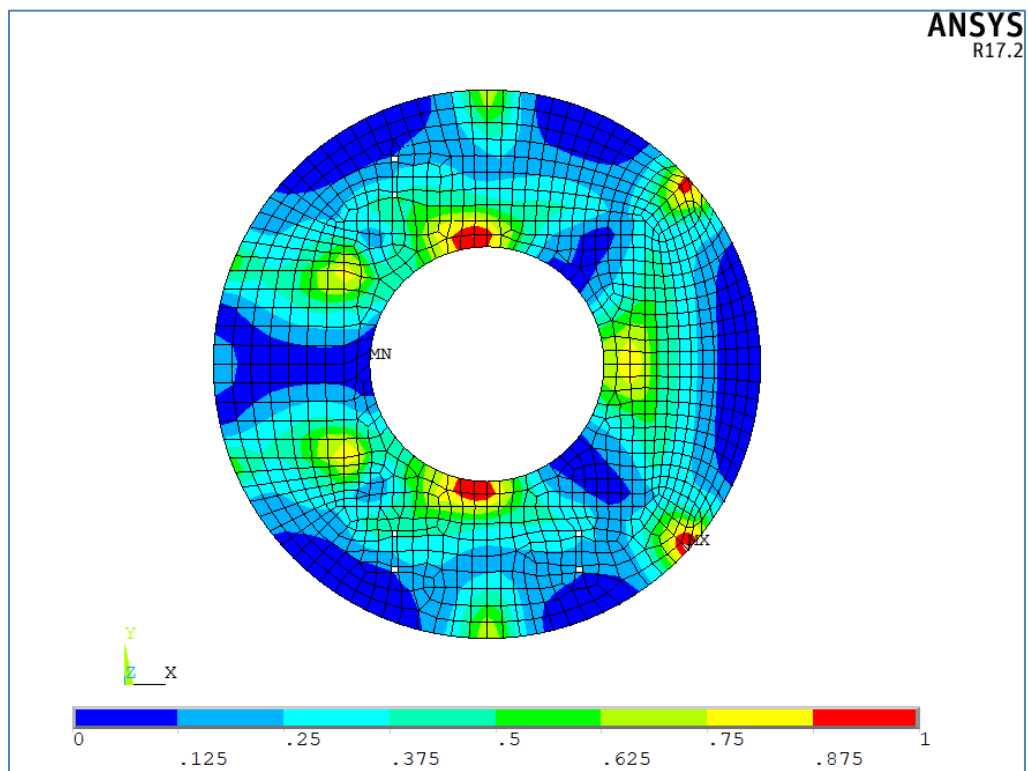


(a) In-Plane Forces and Out-of-Plane Moment Interaction for Concrete Infill

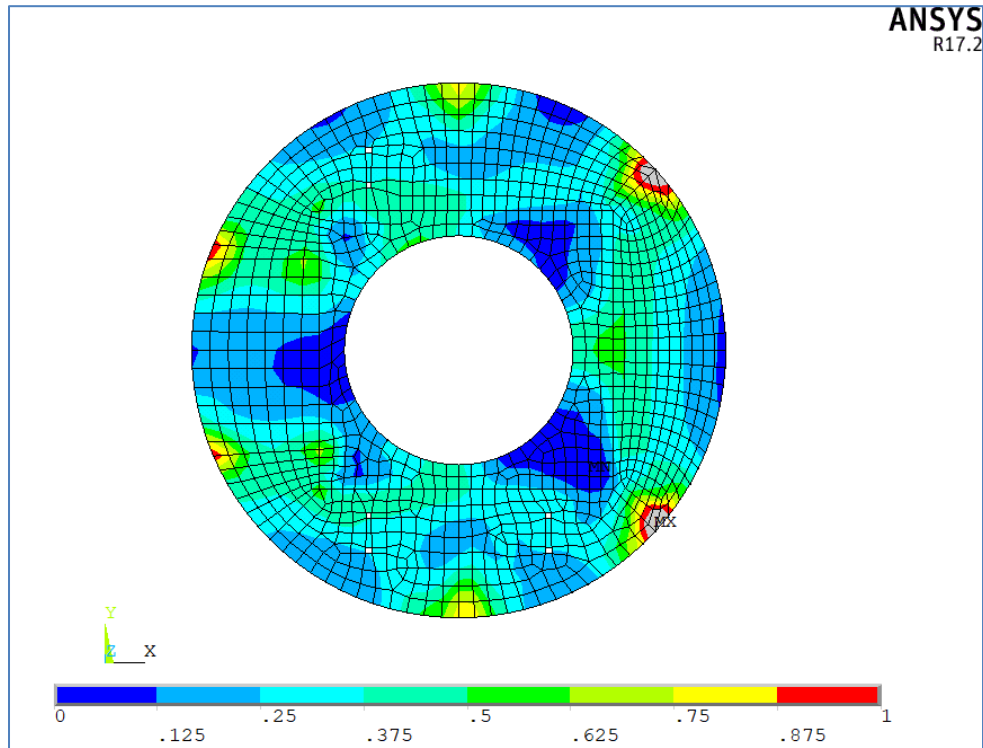
**Figure 9B.10-4: Demand-to-Capacity Ratios for the Inner Mat Foundation Under Abnormal Load Category (SCCV-7)**



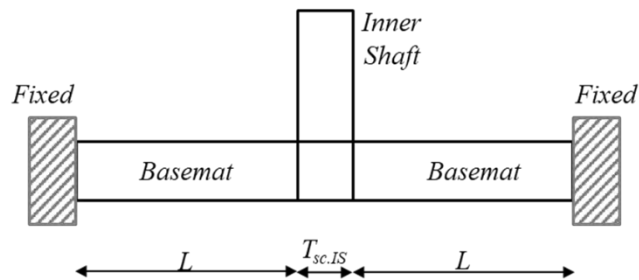
**Figure 9B.10-5: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Wall Under Abnormal Load Category (SCCV-7)**



**Figure 9B.10-6: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Top Slab Under Abnormal Load Category (SCCV-7)**



**Figure 9B.10-7: Demand-to-Capacity Ratios for Out-of-Plane Shear in the SCCV Top Slab Under Abnormal/Extreme Environmental Load Category (SCCV-11)**



**Figure 9B.10-8: Idealized SCCV Wall-to-Mat Foundation Connections**

## **APPENDIX 9B.F – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT INTERNAL STRUCTURES**

### **9B.11 Introduction**

This appendix focuses on the design evaluation of the Steel Bricks™ RPV pedestal as discussed in Subsection 9B.2.2.10.2 and provides bounding estimates of the available margins for the structure.

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendices 9B.B and 9B.D, performed using the one-step approach and the FE models discussed in Appendix 9B.A. The design demands are obtained from bounding design load combinations selected to assess the operational and accidental structural demands on the structure.

#### **9B.11.1 Pedestal Design Methodology**

The design of the RPV pedestal is in accordance with ANSI/AISC N690, similar to the RB structure. Design parameters used in the design evaluation of the RPV pedestal are provided in Table 9B.11-1.

#### **9B.11.2 Bounding Load Combinations and Design Demands**

The evaluation of the RPV pedestal is based on design demands obtained from the bounding load combinations RB-2/3 and RB-9 (See Table 9B-4) selected for the evaluation of the RB structure as discussed in Appendix 9B.G, Subsection 9B.12.2.

#### **9B.11.3 Pedestal Steel Bricks™ Capacity**

The design of the Steel Brick™ modules used in the RPV pedestal satisfies the requirements of ANSI/AISC N690 and the regulatory guidance of U.S. NRC RG 1.243, similar to the RB structure. Refer to Appendix 9B.G, Subsection 9B.12.3 for more details.

#### **9B.11.4 Structural Design Evaluation**

Figure 9B.11-1 and Figure 9B.11-2 present the governing demand-to-capacity ratios for the design of the RPV pedestal for the selected load combinations RB-2/3 and RB-9.

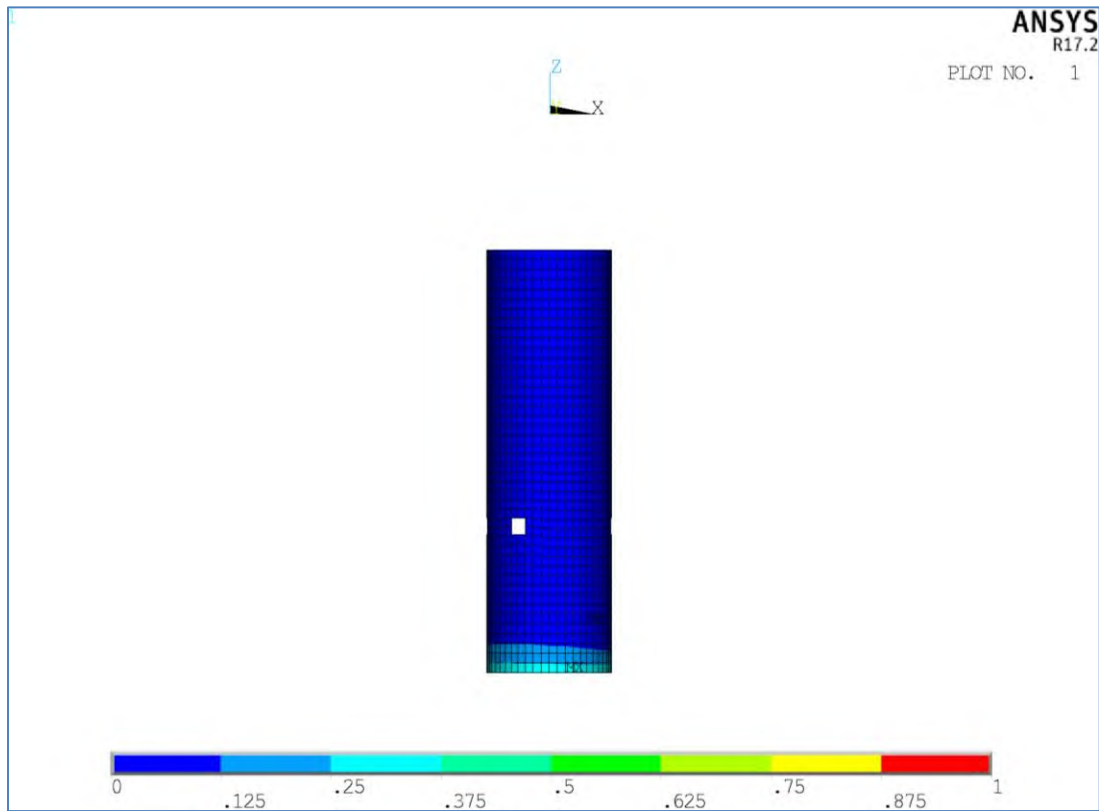
As shown in these figures, the in-plane forces and out-of-plane moment interaction demands in the RPV pedestal under the two load conditions considered are generally small, except for the lower region of the pedestal, where a maximum demand-to-capacity ratio less than 0.8 is noted in Figure 9B.11-2. Higher stress concentrations observed in Figure 9B.11-2 are at the connections with the mat foundation and are, therefore, not relevant for the design of the pedestal.

The 0.8 ratio demonstrates that the design of the RPV pedestal is robust and includes an adequate design margin of 20% in conformance with REGDOC-2.5.2, Section 7.15.1.

**Table 9B.11-1: Design Parameters for Steel Bricks™ in the Reactor Building**

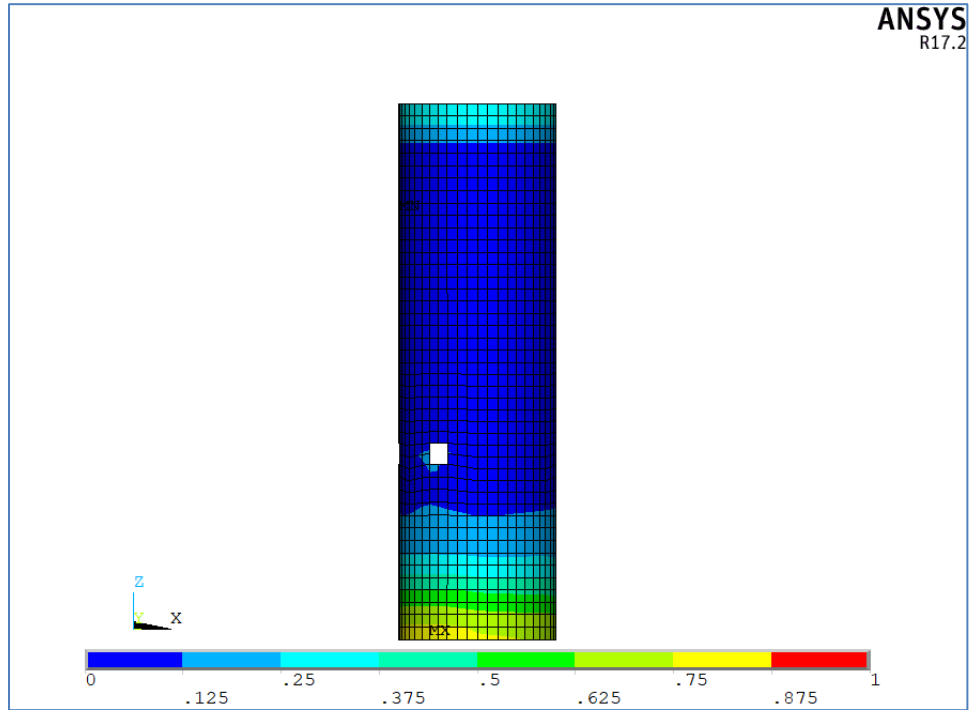
Structural Component	Member Thickness ( $t_{sc}$ ) (mm)	Faceplate Thickness ( $t_p$ ) (mm)	Steel Faceplate Grade	Yield Strength (MPa)	Tensile Strength (MPa)	Diaphragm Spacing (Brick Width) (mm)
RPV Pedestal	970	19.1	ASTM 572 Grade 50	345 <sup>1</sup>	448	970

(1) For Load combinations with accidental thermal loading, a reduced yield strength of 293 MPa is used for the ASTM 572 Grade 50 steel



**Figure 9B.11-1: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RPV Pedestal Under Normal Load Category (RB-2/3)**





**Figure 9B.11-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RPV Pedestal Under Abnormal Load Category (RB-9)**

## **APPENDIX 9B.G – DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE**

### **9B.12 Introduction**

This appendix evaluates the design and provides bounding estimates of the available margins for key structural members of the RB.

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendices 9B.B and 9B.D, performed using the one-step approach and the FE models discussed in Appendix 9B.A. The design demands are obtained from bounding design load combinations selected to assess the operational and accidental structural demands on the structure.

#### **9B.12.1 Design Methodology for the Reactor Building**

The design of the RB structure is in accordance with ANSI/AISC N690 and U.S. NRC RG 1.243. Design parameters used in the design evaluation of the RB Steel Bricks™ are provided in Table 9B.12-1.

#### **9B.12.2 Bounding Load Combinations and Design Demands**

The evaluation of the RB structure is based on results from the bounding load combinations RB-2/3 and RB-9 presented in Table 9B-4. Load combination RB-2/3 is considered bounding for all load combinations for normal load category presented in Table 9B-4. Load combination RB-9 is selected to demonstrate the ability of the integrated RB structures to resist abnormal loading conditions including seismic loadings. This load combination bounds other severe environmental and extreme environmental load combinations listed in Table 9B-4.

Per ANSI/AISC N690, App.N9.2.5, design demands obtained from the various structural analyses may be averaged over panel sections that are no larger than twice the section thickness in length and width. Near openings and penetrations, and in connection regions, the required strength may be calculated by averaging the demand over panel sections no larger than the section thickness in length and width.

Similar to the SCCV, design demands are conservatively presented on element-by-element basis without averaging of the results across adjacent elements as allowed by the ANSI/AISC N690.

#### **9B.12.3 RB Steel Bricks™ Capacity**

The capacities of the RB Steel Bricks™ sections are computed, and the design evaluated per ANSI/AISC N690 in conjunction with ANSI/AISC 360 (Reference 9B.4-52).

Similar to the SCCV, design capacities of the RB Steel Bricks™ sections are determined for each notional half per ANSI/AISC N690, Section N9.3.6b. Strength checks performed for the structure include axial (tension and compression), flexural (out-of-plane), shear (in-plane and out-of-plane) strength checks as well as out-of-plane shear force interaction and in-plane forces and out-of-plane moments interactions.

#### **9B.12.4 Structural Design Evaluation**

The evaluation of the RB structure is performed using the same methodology as for the SCCV described in Appendix 9B.E, Subsection 9B.10.4. Key RB components considered in the evaluation include the RB exterior wall above and below ground, the outer mat foundation and wing walls.

Figure 9B.12-1 through Figure 9B.12-7 present bounding demand-to-capacity ratios for the RB key components for the selected load combinations RB-2/3 and RB-9.

Figure 9B.12-1 presents the governing demand-to-capacity ratio for the design of the outer mat foundation for the normal load category. As shown in this figure, the maximum recorded demand-to-

capacity ratio for the mat's out-of-plane shear is 0.6, with a few stress concentrations observed at the connections of wing walls with the RB and SCCV walls. These stress concentrations are not relevant for the design since they occur entirely within the thickness of the connecting members. As shown in Figure 9B.12-2 and Figure 9B.12-3, the maximum demand-to-capacity ratio for the RB exterior wall below and above grade, respectively, for the same load category is about 0.8 for the in-plane forces and out-of-plane moment interaction. The small stress concentrations observed in Figure 9B.12-3 occur at the connections with the crane supports. The in-plane forces and out-of-plane moment interaction, the most bounding demand for the design of the RB wing walls are generally small, with a maximum demand-to-capacity ratio computed for the structures of 0.3.

For the abnormal load category, Figure 9B.12-4 indicates that the highest demand-to-capacity ratio for the out-of-plane shear, the most governing demand for the design of the outer mat foundation, is less than 0.75. The demand-to-capacity exceedances observed in Figure 9B.12-4 are mainly concentrated around the wing wall junctions and cover an area that is within a distance of 800 mm or less. These exceedances are justified by the modeling approach of element-to-element interaction regions and can be averaged, based on the mat thickness, over a distance up to 1200 mm at connection regions and 2400 mm at other regions. The highest demand-to-capacity ratios computed for the RB wall below and above grade under the abnormal load category are obtained from the in-plane forces and out-of-plane moment interactions shown in Figure 9B.12-5 and Figure 9B.12-6, respectively, and are found to be generally less than 0.8. Demand-to-capacity exceedances observed in Figure 9B.12-6 are at the connections with the crane supports and refuel floor slab. Figure 9B.12-7 also indicates that the in-plane and out-of-plane moment interaction demands on the RB wing walls under the same load category are generally small resulting in a maximum demand-to-design ratio of 0.6 for these structures. Exceedances observed in Figure 9B.12-7 occur at the connection with the mat foundation and SCCV wall and are not relevant to the design since they are within the thickness of connecting elements.

As demonstrated by these evaluation results, the design of the RB structure is robust and includes adequate safety margins in conformance with REGDOC-2.5.2, Section 7.15.1. Demand concentrations observed in Figure 9B.12-1 through Figure 9B.12-7 are either singularities or occur entirely within connecting member thicknesses and can be averaged through the member's thickness and beyond as permitted by ANSI/AISC N690.

### **9B.12.5 Critical Steel Brick™ Connections in the Reactor Building**

The RB exterior wall is connected to wing walls and floor slabs at several elevations and share the same mat foundation with the SCCV as shown in Figure 9B-1. All connections of Steel Bricks™ panels in the RB are rigid except for the semi-rigid connections shown in Figure 9B.6-4. This section provides an overview of methodologies used to model and design the various Steel Bricks™ connections in the RB.

Per ANSI/AISC N690, Appendix N9.4, rigid connections are classified either as full-strength connections with design demand of 125% of the smaller of the corresponding nominal strengths of the connected parts, or as overstrength connections with design demands of 200% of the required strength due to seismic loads plus 100% of the required strength due to non-seismic loads (including thermal loads).

#### **9B.12.5.1 RB Exterior Wall-to-Mat Foundation Connection**

An idealized model is developed for the design of the full-strength rigid RB exterior wall-to-mat foundation connection, referred to as L connection. Figure 9B.12-8 presents the idealized model used for the design of the L connection. As shown in Figure 9B.12-8, the end of the L connection is considered fixed.

Per the full-strength design concept, the weaker of the connected components (RB exterior wall and mat foundation) is expected to fail first. Consequently, design demands and required strength for each individual demand acting on the connection are calculated by identifying the applicable force transfer mechanism and evaluating the potential failure mode or mechanism of the connection.

#### **9B.12.5.2 Wing Wall-to-Mat Foundation Connections**

Overstrength rigid connections are used at the interface between the RB wing walls and the mat foundation to facilitate construction and fabrication. The governing code for the design of this type of connection is ANSI/AISC N690 in conjunction with ANSI/AISC 360. Figure 9B.12-9 presents a schematic for a typical RB wing wall-to-mat foundation connection.

As illustrated in this figure, a steel baseplate is welded to the internal wing wall and is anchored, using cast-in-place steel anchors, to the Steel Bricks™ concrete infill. The design of the overstrength rigid connections at the interface of the wings walls and the mat foundation is governed by the shear strength of the steel anchors, the bearing capacity of concrete and yielding of the base plate.

#### **9B.12.5.3 Wing Walls-to-Floors Connections**

As discussed in Appendix 9B.A, Subsection 9B.6.1.3, semi-rigid connections are used for wing walls-to-floors to reduce the axial and flexural stresses induced due to thermal loadings. Semi-rigid connections between wing walls and floors are identical to floors-to-RB exterior wall connections discussed in Subsection 9B.12.5.4.

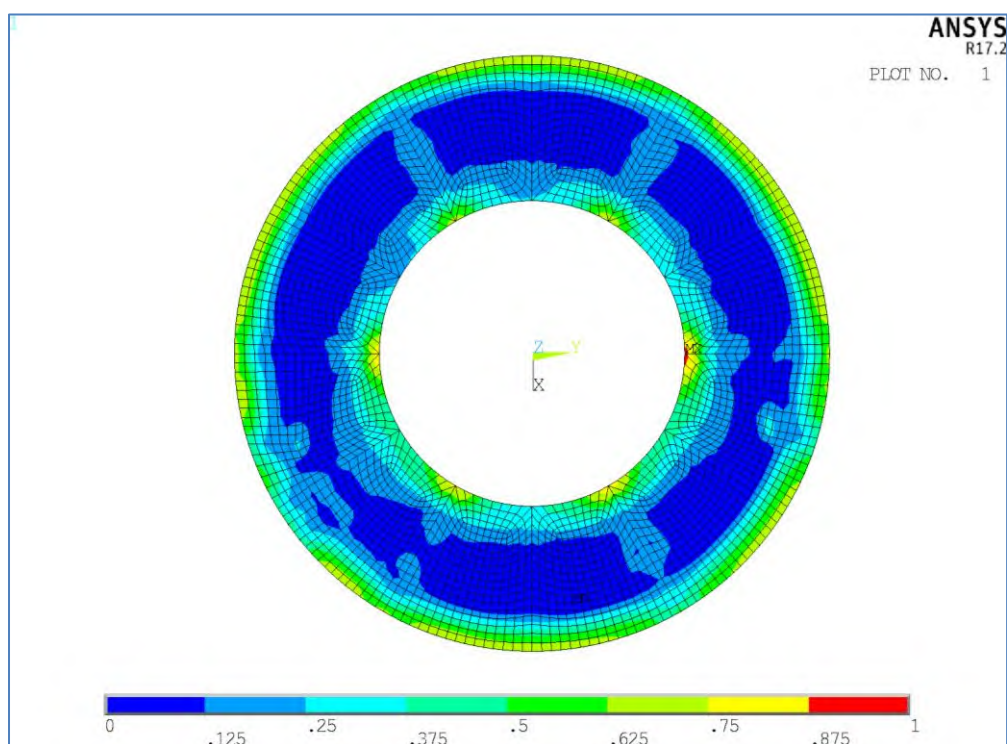
#### **9B.12.5.4 Wing Walls and Floors Connections to RB Exterior Wall**

As discussed in Appendix 9B.A, Subsection 9B.6.1.3, semi-rigid connections are used in wing walls-to-RB exterior wall and floors-to-RB Exterior wall junctions to reduce the axial and flexural stresses due to thermal loadings. Figure 9B.12-10(a) presents typical floor-to-RB exterior wall connection details. The governing code for the design of this type of connection is ANSI/AISC N690 in conjunction with ANSI/AISC 360. As shown in Figure 9B.12-10(a), a stiffened angle cleat is used to connect the floor plates to the outer shaft.

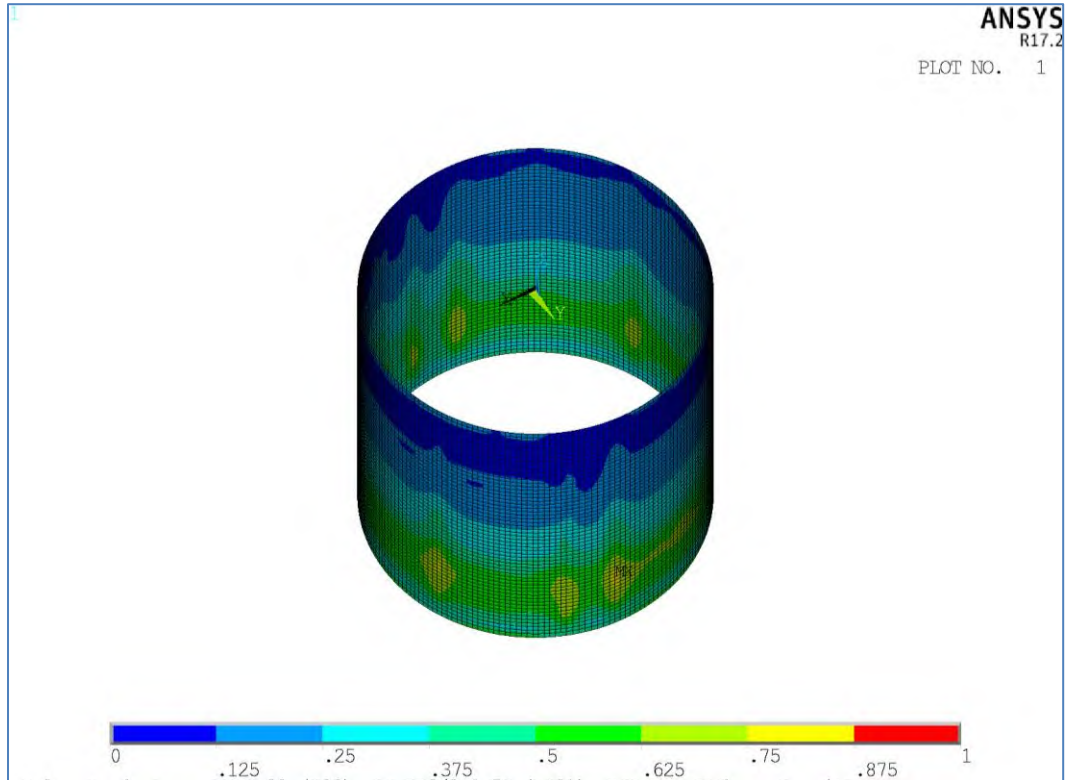
As shown in Figure 9B.12-10(b), Wing walls-to-SCCV wall connections are similar to floors-to-RB exterior wall connections, with vertical angles used in the connections being welded to the outer face plate of the SCCV wall and bolted to wing walls plates. Weld quality assurance, procedure and inspection for the connections to SCCV wall follow ASME requirements as stated in Chapter 3, Subsection 3.5.3.1.1.

**Table 9B.12-1: Design Parameters for Steel Bricks™ in the Reactor Building**

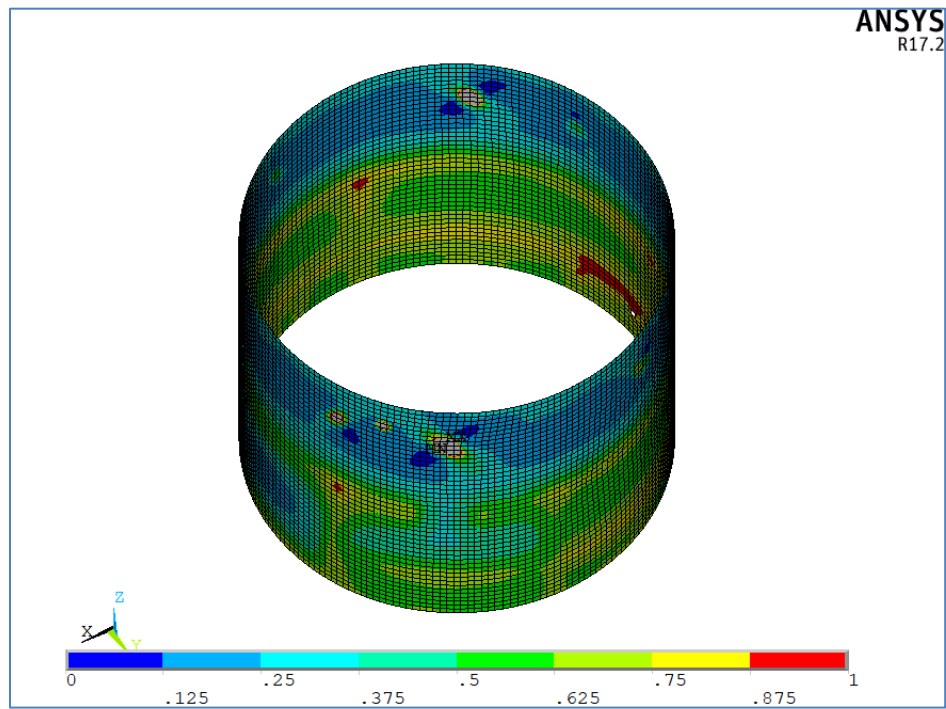
RB Structural Component	Member Thickness (mm)	Faceplate Thickness (mm)	Steel Faceplate Grade	Yield Strength (MPa)	Tensile Strength (MPa)	Diaphragm Spacing (Brick Width) (mm)
RB exterior wall (below grade)	610	10	ASTM 572 Grade 50	345	448	305
RB exterior wall (above grade)	610	15.9				610
Outer mat foundation	1219	15.0				609
Wing walls	457	10				457



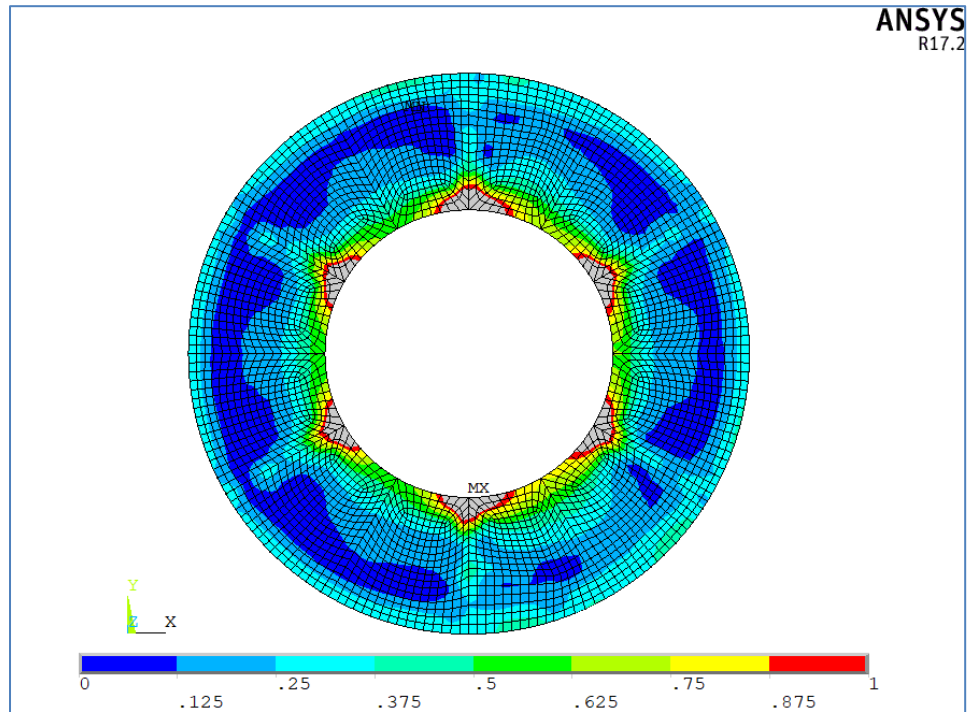
**Figure 9B.12-1: Demand-to-Capacity Ratios for Out-of-Plane Shear in the Outer Mat Foundation Under Normal Load Category (RB-2/3)**



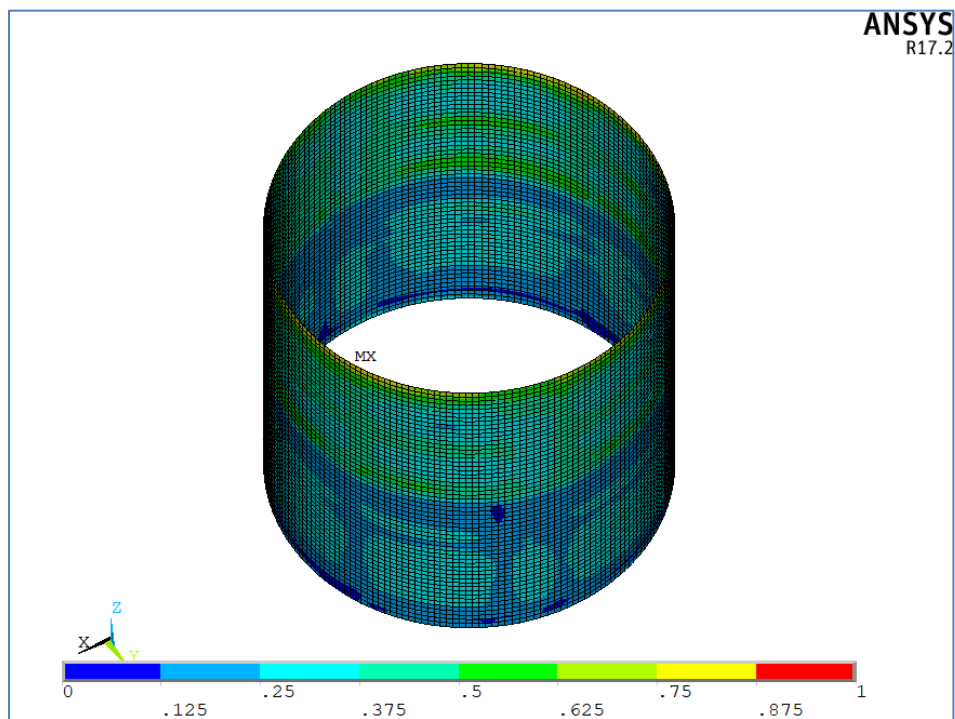
**Figure 9B.12-2: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior wall Below Grade Under Normal Load Category (RB-2/3)**



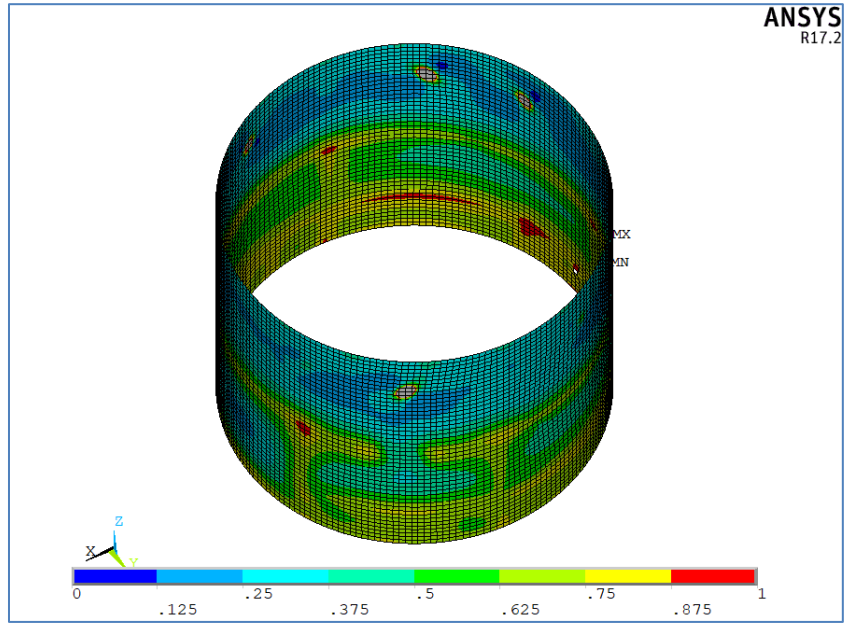
**Figure 9B.12-3: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Above Grade Under Normal Load Category (RB-2/3)**



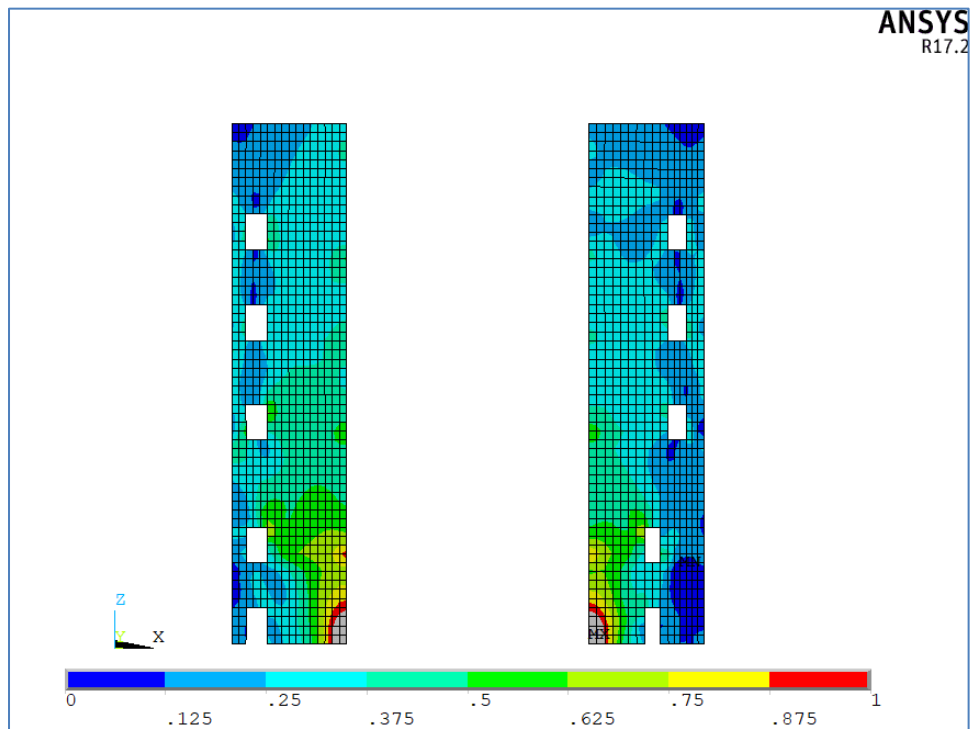
**Figure 9B.12-4: Demand-to-Capacity Ratios for Out-of-Plane Shear in the Outer Mat Foundation Under Abnormal Load Category (RB-9)**



**Figure 9B.12-5: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Below Grade Under Abnormal Load Category (RB-9)**



**Figure 9B.12-6: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for the RB Exterior Wall Above Grade Under Abnormal Load Category (RB-9)**



**Figure 9B.12-7: Demand-to-Capacity Ratios for In-Plane Forces and Out-of-Plane Moment Interaction for Wing Walls Under Abnormal Load Category (RB-9)**



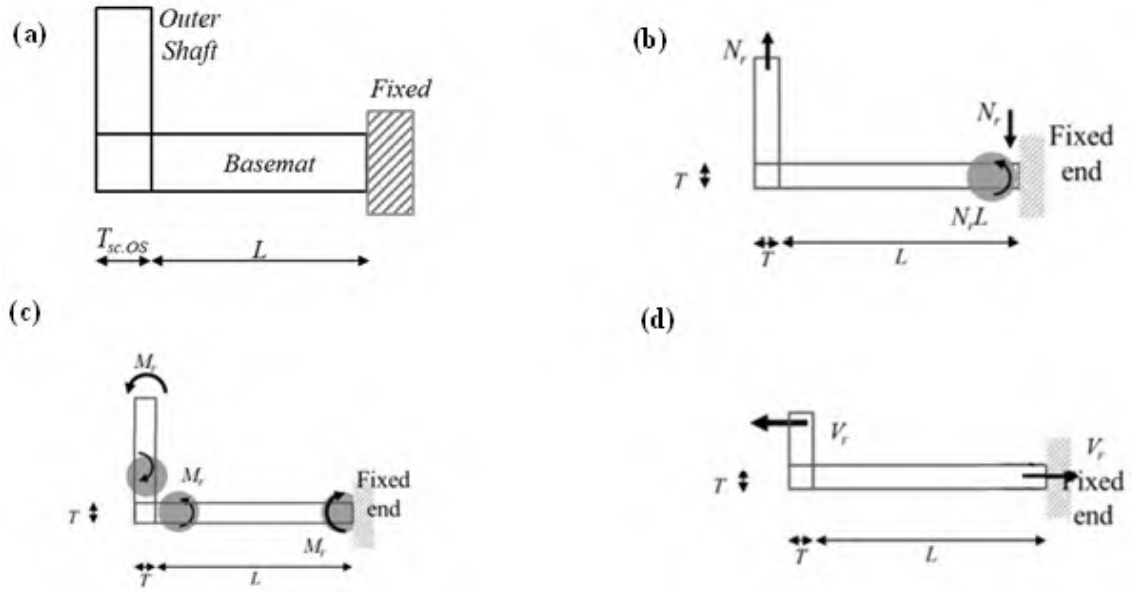


Figure 9B.12-8: Idealized Reactor Building Exterior Wall-to-Mat Foundation Connection

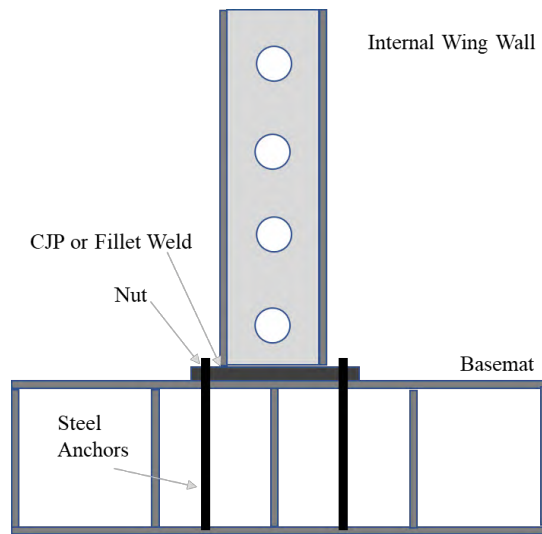
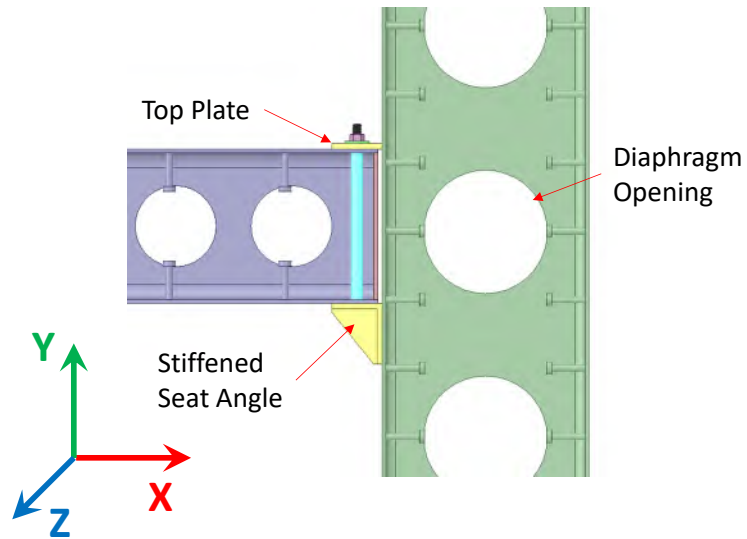
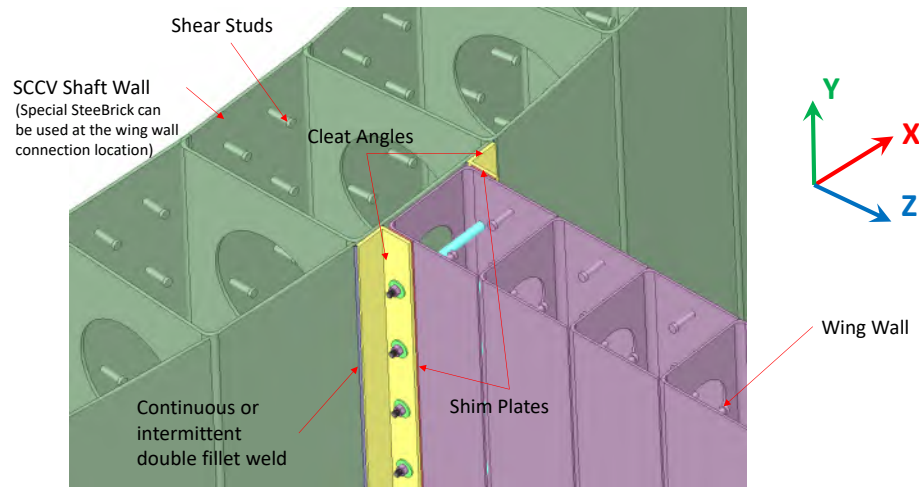


Figure 9B.12-9: Typical Reactor Building Wing Wall-to-Mat Foundation Connection



(a) Typical Floor-to-RB Exterior Wall Connection



(b) Typical Wing Wall-to-SCCV Wall Connection

**Figure 9B.12-10: Typical Wing Wall and Floor Connections to Reactor Building and SCCV Walls**



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33960

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 10  
Steam and Power Conversion Systems**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOV	Air Operated Valve
ASME	American Society of Mechanical Engineers
AWWA	American Water Works Association
BPVC	Boiler and Pressure Vessel Code
BWROG	Boiling Water Reactor Owners' Group
CFD	Condensate Filters and Demineralizers System
CFS	Condensate and Feedwater Heating System
CIV	Containment Isolation Valve
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DC	Direct Current
DCIS	Distributed Control and Information System
DL	Defense Line
EFS	Equipment and Floor Drain System
EHC	Electro-Hydraulic Control
FAC	Flow Accelerated Corrosion
FW	Feedwater

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
GSU	Generator Step Up Transformer
HP	High Pressure
HPU	Hydraulic Power Unit
I&C	Instrumentation and Control
LOOP	Loss-of-Offsite Power
LP	Low Pressure
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MTE	Main Turbine Equipment
NBR	Nuclear Boiler Related
NBS	Nuclear Boiler System
NHS	Normal Heat Sink
PCW	Plant Cooling Water System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
SC	Safety Class
SCN	Non-Safety Class
SDC	Shutdown Cooling System
SIR	Seismic Interface Restraint
SJAE	Steam Jet Air Ejector
SSC	Structures, Systems, and Components
SWM	Solid Waste Management System
TASS	Turbine Auxiliary Steam Subsystem
TB	Turbine Building
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TGCS	Turbine Generator Control System
TGSS	Turbine Gland Seal System
TLOS	Turbine Lube Oil System
TMA	Trip Manifold Assembly
TSV	Turbine Stop Valve



## TABLE OF CONTENTS

<b>10.0 STEAM AND POWER CONVERSION SYSTEMS .....</b>	<b>10-1</b>
10.1 Role and General Description.....	10-1
10.2 Turbine Generator System .....	10-2
10.2.1 System and Equipment Functions .....	10-2
10.2.2 Safety Design Bases .....	10-3
10.2.3 Description .....	10-4
10.2.4 Turbine Integrity.....	10-16
10.2.5 Interfaces with Other Equipment or Systems .....	10-17
10.2.6 System and Equipment Operation .....	10-17
10.2.7 Instrumentation and Control.....	10-20
10.2.8 Monitoring, Inspection, Testing, and Maintenance .....	10-28
10.2.9 Radiological Aspects .....	10-29
10.2.10 Performance and Safety Evaluation.....	10-29
10.2.11 References .....	10-30
10.3 Feedwater Systems.....	10-36
10.3.1 Condensate Filters and Demineralizer System .....	10-36
10.3.2 Condensate and Feedwater Heating System.....	10-47
10.4 Main Steam System .....	10-64
10.4.1 System and Equipment Functions .....	10-64
10.4.2 Safety Design Bases .....	10-64
10.4.3 Description .....	10-64
10.4.4 Materials.....	10-65
10.4.5 Interfaces with Other Equipment or Systems .....	10-65
10.4.6 System and Equipment Operation .....	10-65
10.4.7 Instrumentation and Controls.....	10-66
10.4.8 Monitoring, Inspection, Testing, and Maintenance .....	10-66
10.4.9 Radiological Aspects .....	10-67
10.4.10 Performance and Safety Evaluation.....	10-67
10.4.11 References .....	10-67
10.5 Main Condenser and Auxiliaries System .....	10-71
10.5.1 System and Equipment Functions .....	10-71
10.5.2 Safety Design Bases .....	10-71
10.5.3 Description .....	10-72

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

10.5.4	Materials .....	10-73
10.5.5	Interfaces with Other Equipment or Systems .....	10-74
10.5.6	System and Equipment Operation .....	10-74
10.5.7	Instrumentation and Control.....	10-75
10.5.8	Monitoring, Inspection, Testing and Maintenance .....	10-76
10.5.9	Radiological Aspects .....	10-76
10.5.10	Performance and Safety Evaluation.....	10-76
10.5.11	References .....	10-76
10.6	Moisture Separator Reheater.....	10-80
10.6.1	System and Equipment Functions .....	10-80
10.6.2	Safety Design Bases .....	10-80
10.6.3	Description .....	10-80
10.6.4	Materials.....	10-82
10.6.5	Interfaces with Other Equipment or Systems .....	10-83
10.6.6	System and Equipment Operation .....	10-83
10.6.7	Instrumentation and Control.....	10-84
10.6.8	Monitoring, Inspection, Testing, and Maintenance .....	10-85
10.6.9	Radiological Aspects .....	10-85
10.6.10	Performance and Safety Evaluation.....	10-85
10.6.11	References .....	10-86
10.7	Turbine Auxiliary Systems .....	10-89
10.7.1	System and Equipment Functions .....	10-89
10.7.2	Safety Design Bases .....	10-89
10.7.3	Description .....	10-89
10.7.4	Materials.....	10-91
10.7.5	Interfaces with Other Equipment or Systems .....	10-91
10.7.6	System and Equipment Operation .....	10-91
10.7.7	Instrumentation and Control.....	10-91
10.7.8	Monitoring, Inspection, Testing, and Maintenance .....	10-92
10.7.9	Radiological Aspects .....	10-92
10.7.10	Performance and Safety Evaluation.....	10-92
10.7.11	References .....	10-93
10.8	Circulating Water System .....	10-96
10.8.1	System and Equipment Functions .....	10-96

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

10.8.2	Safety Design Bases .....	10-96
10.8.3	Description .....	10-96
10.8.4	Materials .....	10-98
10.8.5	Interfaces with Other Equipment or Systems .....	10-98
10.8.6	System and Equipment Operation .....	10-98
10.8.7	Instrumentation and Control.....	10-99
10.8.8	Monitoring, Inspection, Testing, and Maintenance .....	10-100
10.8.9	Radiological Aspects .....	10-100
10.8.10	Performance and Safety Evaluation.....	10-100
10.8.11	References .....	10-100
10.9	Generator and Exciter .....	10-104
10.9.1	System and Equipment Functions .....	10-104
10.9.2	Safety Design Bases .....	10-104
10.9.3	Description .....	10-105
10.9.4	Materials .....	10-106
10.9.5	Interfaces with Other Equipment or Systems .....	10-106
10.9.6	System and Equipment Operation .....	10-106
10.9.7	Instrumentation and Control.....	10-107
10.9.8	Monitoring, Inspection, Testing, and Maintenance .....	10-108
10.9.9	Radiological Aspects .....	10-109
10.9.10	Performance and Safety Evaluation.....	10-109
10.9.11	References .....	10-109
10.10	Implementation of Break Preclusion for the Main Steam and Feedwater Lines	10-111

## LIST OF TABLES

Table 10.2-1: Summary of Design Features and Performance Characteristics of the Power Conversion System .....	10-32
Table 10.2-2: Main Turbine Equipment Interfaces .....	10-33
Table 10.3.1-1: Condensate Filter and Demineralizer System .....	10-44
Table 10.3.1-2: Condensate Filter and Demineralizer System Interfaces .....	10-45
Table 10.3.2-1: Condensate Feedwater Heating System Major Components .....	10-56
Table 10.3.2-2: Condensate Feedwater Heating System Interfaces .....	10-57
Table 10.4-1: Main Steam System Subsystem Interfaces .....	10-68
Table 10.5-1: Main Condenser and Auxiliaries System Interfaces .....	10-77
Table 10.6-1: Moisture Separator Reheater System Interfaces .....	10-87
Table 10.7-1: Turbine Bypass System Interfaces .....	10-94
Table 10.8-1: Circulating Water System Interfaces .....	10-101
Table 10.9-1: Generator and Exciter System Interfaces .....	10-110

## LIST OF FIGURES

Figure 10.2-1: Power Cycle Schematic .....	10-35
Figure 10.3.1-1: Condensate Filters and Demineralizer System.....	10-46
Figure 10.3.2-1: Condensate Portion of Condensate and Feedwater Heating System .....	10-60
Figure 10.3.2-2: Feedwater Portion of Condensate and Feedwater Heating System .....	10-61
Figure 10.3.2-3: Vents and Drains for High Pressure Heaters in Condensate and Feedwater Heating System.....	10-62
Figure 10.3.2-4: Vents and Drains for Low Pressure Heaters in Condensate and Feedwater Heating System.....	10-63
Figure 10.4.1-1: Main Steam System .....	10-70
Figure 10.5-1: Main Condenser and Auxiliaries System.....	10-79
Figure 10.6-1: Moisture Separator Reheater System .....	10-88
Figure 10.7-1: Turbine Bypass System Interfaces.....	10-95
Figure 10.8-1: Circulating Water System.....	10-103

## **10.0 STEAM AND POWER CONVERSION SYSTEMS**

### **10.1 Role and General Description**

The components of the power conversion system are designed to produce electrical power utilizing the steam generated by the reactor. This chapter includes the Main Turbine Equipment (MTE) along with the systems listed below.

The MTE is comprised of the following:

- High Pressure (HP) and Low Pressure (LP) Turbines, casings, Main Steam (MS) stop and control valves, non-return valves, reheat steam stop (intermediate stop) and intercept valves, and turning gear
- Turbine Gland Seal Subsystem (TGSS)
- Turbine Lube Oil System (TLOS)
- Extraction Steam System
- Electro-Hydraulic Controls (EHCs) Subsystem

MS piping is not part of the MTE System, refer to Section 10.4 for additional information.

Load rejection capabilities are discussed in Section 10.7.

Other systems described in the sections below include:

- Feedwater (FW) Systems in Section 10.3
- Main Condenser and Auxiliaries (MCA) in Section 10.5
- Moisture Separator Reheater System (MSR) in Section 10.6
- Circulating Water System in Section 10.8

Radiation shielding for personnel protection is provided for all required components as discussed in Chapter 12, Subsection 12.3.2.

## **10.2 Turbine Generator System**

This section describes portions of the MTE, and the Generator and Exciter System is described in Section 10.9. Section 10.4 describes the MS System and Section 10.7 describes Turbine Auxiliary Systems.

The turbine generator consists of a 377 rad/s (3600 rpm) turbine, generator, control valves and associated subsystems.

The turbine has a single flow HP turbine and two double flow LP turbine units in tandem. The generator is direct driven, three-phase, 60 Hz, synchronous generator water cooling system, with an air-cooled stator and rotor.

The TLOS provides 100 percent of the lube oil required by the turbine and generator bearings and turning gear during all modes of operation.

The Extraction Steam System transports extraction steam to the MSR, and FW heaters.

The EHC provides HP hydraulic fluid to actuate the actuators on the main turbine stop and control valves, intermediate stop, and intercept valves, and the Turbine Bypass Valves (TBVs). The EHC subsystem is also used to actuate the trip devices in the trip and overspeed detection system.

### **10.2.1 System and Equipment Functions**

The system and equipment functions associated with the MTE are identified below.

#### **10.2.1.1 Normal Functions (Non-Safety Category)**

The MTE performs the following Non-Safety Category functions:

- Limits the atmospheric air in-leakage through the turbine glands
- TLOS removes water and impurities from the turbine lube oil
- Delivers clean lubricating oil from the lube oil reservoir to the main turbine generator bearings
- The EHC controls rolling the turbine to the rated speed

#### **10.2.1.2 Normal Functions (Safety Category)**

The MTE performs the following Safety Category normal functions.

- Controls Reactor Pressure Vessel (RPV) pressure
- Converts thermal energy contained in the MS and hot reheat steam into rotational torque on a shaft connected to the main generator
- Provides extraction steam from the various stages of the turbine to the first through fifth stages of FW heaters
- Exhausts (extracts) steam to the MSR for reheating (TBD)
- Exhausts steam to the main condenser (from the LP turbine)
- Minimizes radioactive steam leakage from the turbine shaft/casing and valve stems
- Limits atmospheric air in-leakage through the turbine glands
- Delivers clean lubricating oil from the Lube Oil Reservoir to the main turbine generator bearings

- Provides HP hydraulic fluid to actuate the Turbine Stop Valves (TSVs) and TCVs, intermediate stop and intercept valves, and the TBVs

#### **10.2.1.3 Off-Normal Functions (Non-Safety Category)**

This system does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.2.1.4 Off-Normal Functions (Safety Category)**

This system does not perform any Safety Category functions during off-normal conditions.

### **10.2.2 Safety Design Bases**

The MTE is designed to fulfill the safety functions in Subsection 10.2.1.2.

The design of the MTE is such that protection of Safety Category function Structures, Systems, and Components (SSC) from the effects of turbine missiles is afforded as a result of the perpendicular orientation of the turbine to the reactor. The MTE employs redundant turbine overspeed protection system to minimize the probability of turbine missile generation. The favorably oriented turbine generator is located such that the containment and most Safety Category function SSC outside containment are excluded from the low trajectory hazard zone of a potential turbine missile. The turbine is designed to minimize the possibility of failure of a turbine blade or rotor. Turbine integrity is discussed in Subsection 10.2.4. The effects of potential high energy missiles are discussed in Chapter 3, Section 3.5, and Subsection 10.2.4.2.

Safety requirements include the following:

1. Anticipatory scram on fast closure of TCVs/TSVs demand in Mode 1 (Power Operation) is actuated.
2. TCV fast closure on generator load reject in Mode 1 (Power Operation) is actuated.
3. TSV fast closure on main turbine trip in Mode 1(Power Operation) is actuated.
4. Turbine trip on high main condenser pressure in Mode 1 (Power Operation) is actuated.
5. The plant design provides the capability for load following during at least 90% of the plant design life, the plant is capable of a 24-hour load cycle with the following electrical power profiles:
  - a. Starting at 100% power, power ramps down to 50% in two hours
  - b. Power remains at 50% for two to ten hours
  - c. Power then ramps up to 100% in two hours
  - d. Power remains at 100% for the remainder of the 24-hour cycle

The plant design can accommodate a minimum of 20,805 equivalent daily load following cycles.

The main turbine steam flow shutoff characteristics, in collaboration with turbine bypass capacity, and bypass opening response characteristics is designed to prevent a reactor trip from occurring as a result of a load rejection or turbine trip below or at 25% output.

The MS turbine control and protection systems are designed such that a single failure will not result in a main turbine trip.

The main turbine generator system and component designs provide sufficient margin and/or redundancies to account for anticipated abnormal operating conditions and component degradation to avoid limiting conditions of operation over the entire operating cycle.



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

The main turbine is designed to include a corrosion allowance for materials in contact with water or steam systems.

The design of the MTE considers applicable international standards such as compliance with U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.115, (Reference 10.2-2) "Protection Against Turbine Missiles" (Reference 10.2-2) which invokes compliance with 10 CFR 50 Appendix A(4), "Environmental and Dynamic Effects Design Bases" (Reference 10.2-5).

The design of the MTE System meets Canadian Nuclear Safety Commission (CNSC) requirements specified in CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," Section 8.3.3 (Reference 10.2-1), as it relates to providing provisions for overspeed protection and minimization of missiles due to a turbine break-up. Minimization of potential missile impacts is also afforded through the favorable perpendicular orientation of the turbine to the SSC performing safety functions.

### **10.2.3 Description**

MTE design parameters are summarized in Table 10.2-1. A power cycle schematic is presented in Figure 10.2-1.

The turbine is a 3600 rpm, single-shaft, tandem compound, impulse-reaction, two stage reheat, condensing steam turbine. The turbine generator and associated piping, and valves, are located completely within the Turbine Building (TB). Refer to Section 10.4 for information pertaining to the MS system interface with the turbine generator.

The MTE begins at the two MS TSVs and control valves, which admit steam through separate steam leads to two inlet nozzles of the single flow HP turbine. Steam within the HP turbine is extracted for first stage MSR reheating and for fifth stage FW heating. HP turbine exhaust steam is directed to the MSR (Subsection 10.6.2) where the steam is dried and goes through two stages of reheating before entering the two LP turbines. Some HP turbine exhaust steam is also routed to the fourth stage FW heater. Steam within each of the two LP turbines is extracted for the first, second and third stages of FW heating. Exhaust steam from the LP turbines is directed to the condenser.

The main turbine is equipped with a turning gear which is used to rotate the turbine generator shafts slowly and continuously, as needed when the main turbine is not in-service, such as the early part of turbine warming and after the rotor has stopped at shutdown.

The system's physical layout provides protection to essential systems and components, as required, from the effects of high and moderate energy turbine generator system piping failures or failure of the connection(s) from the LP turbine exhaust hoods to the condenser. Essential systems are also protected from turbine missiles due to overspeed events. Failure of turbine generator equipment does not preclude safe shutdown of the reactor systems.

The design and control of the turbine valves is such that there is redundancy from the input device to the signal going back out to the three servo valves, with the first possible single failure point being the mechanical portion of the servo valve itself. The servo is biased to fail closed on loss of all electrical signals and on failure to drain on loss of servo valve jet lube oil flow.

The turbine generator is oriented within the TB to minimize the potential of turbine missiles damaging any Safety Category function equipment or structures. Favorably oriented turbine generators are located such that the containment and Safety Category function SSC outside containment are minimized from the low trajectory hazard zone (See Subsection 10.2.4.2 for discussion of turbine missiles).

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Material and equipment selection for the MTE components is based on a 60-year design life, with appropriate provisions for maintenance and replacement.

The MTE is designed for an operating life based on continuous plant operation at full power for a 24-month cycle, including a refueling outage of approximately fourteen (14) days.

#### **10.2.3.1 Turbine Gland Seal Subsystem**

The TGSS components are designed to Category Non-Nuclear Seismic.

The TGSS supplies sealing steam to the turbine shaft/casing to prevent the escape of radioactive steam and to prevent air in-leakage through sub atmospheric turbine glands.

The TGSS takes sealing steam from the Turbine Auxiliary Steam Subsystem (TASS) only in low power conditions and during full power the sealing steam comes from the seals at the HP side of the HP turbine. The TASS takes steam from the MS equalizing header. The Steam Jet Air Ejectors (SJAEs) and offgas take steam from the TASS.

#### **10.2.3.2 Turbine Lube Oil System Description**

The TLOS supplies the lube oil to the turbine, generator, and exciter bush bearings. The TLOS skid includes redundant pumps to continuously supply oil to the bearings to protect against commercial losses. The lube oil skid also contains storage tank and heat exchangers which support the TLOS. An oil conditioning system is provided with TLOS.

#### **10.2.3.3 Extraction Steam Subsystem**

The purpose of the Extraction Steam System is to provide the means to transport extraction steam from the steam turbine to the MSR first stage reheater and FW heaters for regenerative FW heating and protect the steam turbine against overspeed and water induction conditions.

Exhaust steam from the HP turbine is directed to the FW Heater No. 4 and the moisture separator input of the MSR, the extraction steam from the HP turbine is directed to the FW Heater No. 5 and 1st stage reheater of the MSR. Extraction steam from the LP turbines provides heating steam to the first, second, and third stage FW heaters.

#### **10.2.3.4 Electro-Hydraulic Controls**

The EHC subsystem provides HP hydraulic fluid to position the TSVs and TCVs, intermediate stop and intercept valves, and the TBVs. The turbine hydraulic system is also used to actuate the trip devices in the trip and overspeed protection circuits for the turbine.

The EHC system tubing and components are designed to Category Non-Nuclear Seismic.

#### **10.2.3.5 Component Description**

The following defines the requirements specific to the major equipment items in the MTE System.

##### **10.2.3.5.1 Main Steam Turbine Stop and Control Valves**

Two hydraulically operated MS turbine stop, and control valves (four valves total) admit steam to the HP turbine. The primary function of the MS TSVs is to isolate the MS from the turbine and quickly shut off the steam flow to the turbine under emergency conditions. The primary function of the control valves is to control steam flow to the turbine in response to the Turbine Generator Control System (TGCS).

The MS turbine stop and control valves and their supports are designed such that excessive vibration cannot result from excitation caused by turbine operating harmonics and hydraulic instabilities in the valves or steam flows.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Limit switches and valve position logic indication used on these valves are based on proven experience and satisfy triple redundancy criteria. Leak tight enclosures are provided for local turbine control and test logic limit switches and field termination junction boxes.

Valves which incorporate a pilot valve are designed based on proven experience.

The MS TSVs are hydraulically operated by HP fire-resistant hydraulic fluid in an open-closed mode either by the turbine overspeed protection in response to a turbine trip signal, fast-acting valve for tripping, or by a small solenoid valve for testing. The disks are unbalanced and cannot open against full differential pressure. A bypass is provided to pressurize the below seat areas of the two valves and supply steam for turbine casing and steam chest warming. Springs provided with the valves are designed to close the MS turbine control valves under emergency conditions.

The stop valves provide a method to allow prewarming during turbine startup.

Each MS turbine stop valve contains a steam strainer and integral screens to prevent foreign matter from entering the MS turbine control valves and turbine.

The control valves are designed to provide steam shutoff adequate for turbine speed control. The valves are of sufficient size, relative to their cracking pressure, to require partial balancing.

Each control valve is hydraulically operated by HP fire-resistant hydraulic fluid supplied through a servo valve. The control valves are designed to close in 2.5 seconds and fast-close in 0.08 seconds.

The turbine control valves are capable of full stroke opening and closing times with the reactor at 100% power, including load following.

The turbine control valves are designed to withstand extended periods of operation at less than full load, including load following.

The MS turbine stop and control valves will not produce unacceptable stress or reactions on the bearings during testing. Valve testing does not result in bearing temperature alarms or vibration alarms.

The MS turbine stop and control valves are designed to be compatible with control functions of the Reactor Pressure Control (RPC).

#### **10.2.3.5.2 Intermediate Stop and Intercept Valves**

One hydraulically operated reheat Intermediate Stop and Intercept Valve is provided for each LP turbine. The valves are located as close to the LP turbine as required to control turbine speed and protect the turbine against overspeed from steam and water energy stored between the MS stop and control valves and the intermediate stop and intercept valves.

The reheat intermediate stop and intercept valves are designed to close rapidly to control turbine overspeed. Upon loss of load, the valves first close then throttle steam to the LP turbine as required to control speed. The valves close on a turbine trip or if the valves fail to operate properly to control overspeed. Each valve is capable of opening against full system pressure and capable of opening against a pressure differential of approximately 15 percent of the maximum expected system pressure.

The reheat intermediate stop and intercept valves and their supports are designed such that excessive vibration cannot result from excitation caused by turbine operating harmonics and hydraulic instabilities in the valves or steam flows.

Valve operating mechanisms are of a fail-safe (i.e., fail closed) design that minimizes the use of complicated valve linkages and uses components that are based on proven experience.

Limit switches and valve position indication used on these valves is based on proven experience and satisfy the triple redundancy criteria. Leak tight enclosures are provided for local turbine control and test logic limit switches and field termination junction boxes. Refer to Chapter 7 for information pertaining to triple redundancy.

Intermediate stop valves which incorporate a pilot valve are designed based on proven experience.

#### **10.2.3.5.3 Main Turbine (High and Low Pressure)**

The turbine is capable of startup from cold conditions to full load within twelve hours, including rotor preheating.

The turbine is designed to separate water from the steam and internally drain it to the next lowest extraction point.

The HP turbine receives steam through two steam leads, one from each control valve outlet. The steam is expanded axially across several stages of stationary and moving blades. Extraction steam from the HP turbine is used to supply the fifth stage of FW heating and MSR first stage reheater heating steam. HP turbine exhaust steam is collected in a cold reheat pipe and routed to the MSR inlet.

Two units of LP turbines coupled together with the HP turbine shaft convert thermal energy into mechanical rotational energy to drive the generator. Each LP turbine receives hot reheat steam from its connected MSR outlet through the intermediate stop and intercept valves. Each LP turbine is a double flow with six extraction points for FW heating. Each LP turbine exhausts directly to the main condenser connected underneath each LP turbine.

#### **10.2.3.5.4 Turning Gear**

The main turbine is equipped with a turning gear. The primary function of the turning gear is to rotate the turbine generator shafts slowly and continuously, as needed during shutdown periods. Use of the turning gear during startup eliminates the necessity of "breaking away" the turbine generator from standstill with steam, and thereby provides a more uniform and controlled startup. The turning gear may also be used to rotate the rotor incrementally at desired intervals for inspection and also prevents rotor bowing.

Interlocks preventing turning gear operation under conditions which may cause equipment damage, such as loss of lubricating or lift oil, are included in the design of the main turbine.

The turning gear has the ability to engage and start upon zero turbine rotor speed and sufficient lube oil pressure and disengage at startup.

#### **10.2.3.5.5 Extraction Steam Non-return Valves**

The extraction check valves are installed in the appropriate turbine extraction lines supplying steam to FW heaters. The valves prevent backflow of steam from FW heaters to the main turbine after sudden load reductions or turbine trip. The disks of the extraction non-return valves are furnished with air operators allowing them to operate freely when the cylinders are supplied with instrument air pressure controlled by the turbine control system. Upon loss of air pressure or upon equalization of the air pressure on both sides of the piston, the cylinder springs and associated linkage initiate and assist in closure of the valves.

#### **10.2.3.5.6 Piping and Valves**

The turbine system piping includes the MS connections from the MS control valves to the HP turbine steam inlet nozzles, and HP turbine and LP turbine exhaust hoods instrumentation piping.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

The MS control valves are mounted on the HP turbine steam inlet nozzles.

Drain piping is provided for each turbine drain location. The drain piping is large enough to ensure an adequate flow area for the maximum amount of water to be handled under any operating condition and minimal pressure differential conditions. Continuous bleed type bypass is used in areas where water can collect during operation.

The flow area of the main turbine piping provides an acceptable steam velocity based on successful operating experience, considering expected fluid conditions (pressure, temperature, and moisture levels) and considering the specified material requirements.

The turbine system piping is designed to the requirements of American Society of Mechanical Engineers (ASME) B31.1, "Power Piping" (Reference 10.2-3).

Pipe thicknesses are calculated with the appropriate erosion-corrosion allowance to preclude failure during design life.

Double isolation valves are provided, where necessary or where required by code, to facilitate online maintenance of HP/high temperature systems.

Wherever possible, long radius bends (1.5 times nominal pipe size) are used. Piping layouts that result in 90° elbows, miters, etc. are also minimized.

#### **10.2.3.6 Turbine Gland Seal Steam System**

The TGSS is capable of providing sealing steam to and from all HP and LP turbine gland seals.

The TGSS is designed to provide sealing steam to the HP turbine shaft seals in a range of turbine shell pressures from full vacuum to full-load operating pressure. The LP turbine shaft seals normally operate against a vacuum.

The TGSS has two independent sources of steam:

- The HP turbine packing leakoff supply steam, which is available to TGSS during normal and transient operating conditions
- The Main Steam Line (MSL) during startup operation condition

Means for automatically and manually controlling gland steam are provided to maintain proper gland steam conditions during normal and transient operating conditions.

The turbine seals are also supplied with a partial vacuum maintained by the Gland Steam Condenser (GSC) using redundant exhaust blowers to prevent steam leakage in TB atmosphere. The GSC is designed to prevent the entrainment of air in the gland steam condensate.

TGSS piping is designed to meet the requirement of ASME B31.1 (Reference 10.2-3) as applicable.

Corrosion/erosion resistant materials are used for components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of corrosion/erosion resistance of the material is consistent with the temperature, moisture content and velocity of the wet steam to which the component is exposed.

Relief valves on the seal steam header prevent excessive seal steam pressure, which protects the system from malfunction and mis-operation. The relief valves limit the steam seal header pressure to a safe level.

#### **10.2.3.7 Gland Steam Condenser**

The GSC is a shell and tube heat exchanger designed to condense the maximum leakoff steam as required, contained within the air-steam mixture withdrawn from the HP and LP turbine gland seals and the stem leak-offs of the TSVs, TCVs, intermediate stop and intercept valves, and TBVs as required. The air-steam mixture is drawn into the GSC where steam is condensed on the outside of the tubes and air and non-condensables exit to the GSC exhaust blowers. The condensed steam is drained through a trap to the main condenser. Heat is rejected to the condensate flowing through the GSC tubes.

The outer leakoff annulus region of the glands of the turbine are connected to the GSC.

The GSC drain piping is routed and continuously pitched downward to the main condenser. The drain piping is provided with a loop seal to isolate the GSC from the lower pressure of the main condenser. The GSC loop seal is installed with a telltale. The telltale prevents water from rising in the GSC due to a GSC tube rupture. This excludes the possibility of flooding the GSC, adversely impacting the performance of the exhaust fan and the suction line (leakoff) at the turbine gland seal. The overflow is directed toward the nearest sump of the Equipment and Floor Drain System (EFS) (Chapter 9A, Subsection 9A.9.3). The loop seal is filled with demineralized water before the GSC exhaust blower is placed in operation. The GSC is mounted on a pedestal, or an opening provided to accommodate the loop seal below the GSC body.

The GSC is a constant flow type, which means the condenser is located in the Condensate and FW Heating System (CFS) (Subsection 10.3.2) auxiliary condensate loop so that it is supplied with a constant amount of condensate during all operation modes. To maintain condensate flow through the GSC, the water box is supplied with a loop seal to limit the flow to the condenser during operation.

#### **10.2.3.8 Gland Steam Condenser Exhaust Blowers**

Two 100% capacity Alternating Current (AC) motor-driven GSC exhaust blowers maintain a slight vacuum on the GSC shell to remove non-condensable gases. The GSC exhaust blowers are centrifugal type.

Non-condensable gases are removed from the GSC and exhausted to the stack by two parallel, redundant, 100% GSC exhaust blowers. The GSC exhaust blower suction piping pitches downward away from the blower to a low point drain before entering the fan. The low point drain is drained to the main condenser and includes a trap, which prevents condensate from running back to the blower from the vent piping and prevents air from being drawn into the main condenser.

The GSC exhaust blowers have a manual butterfly valve at the inlet and a free-swing check valve in the discharge lines. The check valves prevent recirculation of air through the idle blower when the other is running.

#### **10.2.3.9 Turbine Gland Seal Steam System Piping and Valves**

The TGSS includes an air-operated pressure regulating feed valve for the TASS.

The TGSS piping is designed to have sufficient flexibility for expansion of turbine parts due to thermal growth.

A continuous orifice drain to the main condenser is located on the inlet side of the steam seal feed valve extraction, and steam seal feed valve to avoid accumulation of water at the feed valves.

The steam seal feed valve has a bypass which is located at the highest point in its pipeline.

TGSS includes an air-operated seal steam dump valve for the steam seal header. the seal steam dump valve acts as unloading valve when, at higher turbine loads, more steam is leaking from the pressure packing than is required by the vacuum packing.

The seal steam dump valve has a bypass.

The piping to the turbine shaft packing connections is run from the steam seal header piping, between the feed and dump valves. The steam seal header incorporates a minimum of one low point drain. The steam seal header is drained to the main condenser via this low point drain utilizing a continuous drain to prevent the buildup of water in the steam seal header during normal operation.

The TGSS includes test connections to permit performance testing.

#### **10.2.3.10 Turbine Lube Oil System**

The TLOS provides 100% of the oil required by the turbine generator bearings during turning gear, startup, shutdown, normal, and emergency operating conditions. Sufficient flow margin is provided to account for anticipated off design operating conditions and wear.

The TLOS contains lube oil primary and secondary pumps, an emergency Direct Current (DC) powered pump, a lube oil heat exchanger, and a lube oil conditioner system that removes water and impurities from the oil.

A full-flow filtration system downstream is provided to meet the filtration requirements. The system is provided with the capability for cleaning the filters while online without interrupting oil flow by switching from one cooler/filter assembly to the other.

To minimize corrosion due to moisture contamination, a lube oil conditioner system with water removal capability is provided.

TLOS oil coolers and heaters maintain supply oil temperature within operating limits. A high temperature alarm functions during all design operating conditions.

In the event that the Primary and Secondary AC Bearing Oil Pumps are inoperative due to AC power loss, or a turbine trip due to loss of lube oil pressure, the DC motor-driven Emergency Bearing Oil Pump is sufficiently sized and automatically starts to ensure the lube oil is supplied to the bearings continuously during rotor coast down to prevent damage. Manual controls for this pump are also provided.

Primary and Secondary Lift Oil Pumps are provided for the turbine and generator bearings for turning gear operation as required.

The top of the Lube Oil Reservoir is provided with gravity-closed doors with seals to prevent contamination.

A means for removal of water from the lube oil at the end of an outage is provided to prevent water contamination.

Provisions for the flushing of the Lube Oil Reservoir, supply lines, and return lines is provided to ensure acceptable contaminant levels for operation.

Provisions for heating the lube oil during flushing are provided on the Temporary Flushing Skid.

The Secondary Lube Oil Vapor Extractor is powered from a different bus than the Primary Lube Oil Vapor Extractor.

Provisions for obtaining representative samples of lube oil are included to ensure requisite quality is maintained.

#### **10.2.3.11 Lube Oil Pumps**

TLOS Primary and Secondary AC Bearing Oil Pump capacities are identical and are sufficient to maintain oil pressure and adequately lubricate the shaft bearing, and turning gear during startup, shutdown, and normal operation.

In the event of either pump failure or maintenance, pumps can be used interchangeably for supplying the TLOS during all modes of operation. The discharge of the Primary and Secondary AC Bearing Oil Pumps enters the Lube Oil Coolers first, then through the full-flow filters, and then is supplied to the bearing header. The Primary and Secondary AC Bearing Oil Pumps are self-venting to prevent air from becoming trapped in the piping.

The DC Emergency Bearing Oil Pump is a DC motor-driven pump that prevents the loss of bearing oil flow by providing emergency backup upon loss of AC power.

Primary and Secondary Lift Pumps are AC motor driven, positive displacement pumps that supply lube oil to the rotor bearings when the turbine is on turning gear.

The Secondary AC Bearing Oil Pump motor is powered from a different bus than the Primary AC Bearing Oil Pump motor.

The power to the Primary and Secondary AC Bearing Oil Pumps is independent and different from that used to control and power the DC Emergency Bearing Oil Pump. This ensures that a failure of a single power source does not affect both the Primary and Secondary AC Bearing Oil Pumps and the DC Emergency Bearing Oil Pump.

The AC and DC motor-driven oil pumps are continuously vented to remove entrained air and the vent line is of sufficient size to prevent plugging.

#### **10.2.3.12 Lube Oil Reservoir Skid**

The Lube Oil Reservoir Skid contains a reservoir with sufficient capacity to contain all oil required by the system for normal operation, turbine trip and to provide submergence of the pumps located in the reservoir. The capacity of the reservoir results in considerable oil above the pump inlets and a low recirculation rate with positive time for air detrainment.

The Lube Oil Reservoir Skid is located below the turbine operating floor so that all oil drains back to the reservoir by gravity.

#### **10.2.3.13 Lube Oil Filters**

The filtration system consists of two 100% capacity in-line full-flow duplex filters. Contamination or plugging of the filter element in-service requires the backup filter element be manually brought into service. Connected to the filters, an oil vent with orifice to continuously drain lube oil back to the reservoir is provided.

#### **10.2.3.14 Turbine Lube Oil System Piping and Valves**

TLOS supply and drain piping is comprised of all welded construction to the maximum extent possible to eliminate possible leakage pathways such as threaded and flange connections.

Pressurized oil lines in the vicinity of hot parts of the turbine are enclosed in a guard line to the maximum extent possible. The guard lines are designed such that in the event of a ruptured pressure line, the pressurized oil drains into the guarding pipe and returns to the reservoir. Pressurized oil lines not in the vicinity of hot parts of the turbine are not required to be enclosed in guarding lines.

TLOS piping downstream of the full-flow filters are corrosion resistant stainless steel, to prevent contamination of lube oil due to pipe corrosion.



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

With the Lube Oil Vapor Extractor maintaining a negative pressure in the Lube Oil Reservoir, this arrangement assists in holding a slight negative pressure within all drain enclosures, thereby, preventing oil and vapor leakage from drain piping.

A separate vent for TLOS is provided. Location of the vent is such as to prevent accidental contact of discharge vapors and gases with fire or sparks. The vent is located away from air intakes for facilities or equipment. TLOS vent is provided with a high efficiency mist eliminator and dual vapor extractors, packaged in a single unit mounted on top of the tank.

A Cooler/Filter Selector Valve is used to switch flow from one oil cooler/filter to the standby combination. The Cooler/Filter Selector Valve assembly allows for full volume flow while switching to the alternate cooler/filter.

The Bearing Pressure Regulator Valve self-regulates the bearing header pressure. This valve has an external orifice which provides minimum necessary oil flow when the valve is not in-service. Adjustments and maintenance are performed through an access cover in the tank top.

The Vapor Extractor Inlet Valve is a manual valve that regulates the amount of vacuum created in the tank by the Vapor Extractor. This valve is accessible from the top of the reservoir.

Oil drain lines pitch downward toward the Lube Oil Reservoir.

#### **10.2.3.15 Extraction Steam System Piping and Valves**

Flow Accelerated Corrosion (FAC) resistant materials are used for all components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of FAC resistance of the material is consistent with the temperature, moisture content and velocity of the wet steam to which the component is exposed.

Extraction drains consist of a drain pot or orifice and a drain line.

Extraction motor-operated isolation valves are provided to prevent water induction into the turbine. The extraction isolation valves are designed to permit a functional test during plant operation without tripping the turbine.

The extraction air-assisted non-return valves limit contributions to turbine overspeed from steam and water in the extraction lines and FW heaters in the event of a turbine generator trip signal. The air-assisted non-return check valve air pistons go through full stroke in the closing direction within two seconds of tripping the air relay dump valve.

Non-return valves are located five pipe diameters downstream from flow disturbances, such as tees and elbows, and ten pipe diameters downstream from control valves, in accordance with EPRI-NP-5479, "Application Guide for Check Valves in Nuclear Power Plants" (Reference 10.2-4).

The Extraction Steam System, where required, has a motor-operated isolation valve and an air-assisted non-return valve located in extraction line from the turbine to the FW heaters to prevent water induction backflow and overspeed of the turbine.

The design of the extraction piping, valves, drains and associated controls are in accordance with ASTM TDP-2, "Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants" (Reference 10.2-6).

A drain is located at each low point in the extraction steam piping system where water may collect during startup, shutdown, or normal operation. For extraction lines with isolation valves located at an elevation higher than its associated FW heater, the extraction piping between the isolation valve and the FW heater is sloped to eliminate possible low points.

#### **10.2.3.16      Electro-Hydraulic Control System**

The EHC system consists of hydraulic fluid, Hydraulic Power Unit (HPU), Trip Manifold Assembly (TMA), Fluid Actuator Supply tubing, Fluid Actuator Drain tubing, Emergency Trip System tubing, and associated tubing, valves, and instrumentation.

The HPU consists of a fluid reservoir, redundant pumping systems, fluid coolers, accumulators, space heaters, air dryer, and a fluid transfer and filtering unit (transfer and Fuller's earth filtration unit). The transfer and Fuller's earth filtration unit consists of a transfer pump, Fuller's earth filters, backup filter, and associated valves. The HPU Fluid Actuator Supply for the TSVs, TCVs, and intermediate stop and intercept valves is capable of isolation from the fluid actuator supply for the TBVs. The HPU is located at a low elevation. The HPU is located such that all hydraulic Fluid Actuator Drain lines pitch downward.

The TMA consists of two parallel triple redundant circuits (Circuits A and B). Each TMA circuit consists of three Electronic Trip Device solenoid valves. Refer to Chapter 7 for information pertaining to triple redundancy. The TMA also includes (total for both circuits) six pairs of isolation valves, a continuous-flow transfer valve assembly, and two air relay dump valves. The TMA is located on the HPU skid.

The EHC system includes in-line full-flow filters for the control of the particulate contaminants in supply lines to ensure 100% of the hydraulic fluid is filtered. Parallel filters and valving are provided to allow cleaning of a filter without interrupting fluid flow while online.

The HPU coolers are water-cooled shell and tube heat exchangers.

The EHC system includes means for the control of hydraulic fluid chemistry and removal of moisture from the hydraulic fluid to prevent corrosion of the system which contaminates the fluid. Provisions for sampling the hydraulic fluid is provided to enable the verification of chemistry and moisture content.

The EHC system design includes redundant instrumentation with a coincident logic trip system to improve system reliability by reducing false and spurious trips.

Each trip initiating signal from the EHC system is monitored to identify the source of the trip signal.

The EHC system provides an independent and redundant backup electrical overspeed trip circuit, to quickly close the TSVs, TCVs, intermediate stop and intercept valves, to quickly shut down the turbine in the event of an unsafe condition. The EHC system is designed to minimize false and spurious trips during normal operation and testing of the trip system.

The EHC system is provided with redundancies such that a failure or malfunction of a single component or power source does not result in an unsafe overspeed or trip actuation of the turbine.

The EHC system includes provisions for online testing of trip devices to verify the system is operable and will function correctly.

The EHC system supports the hydraulic controllers for the TSVs, TCVs, intermediate stop and intercept valves, and TBVs that close or open the valves (as applicable) sufficiently after receipt of a trip signal to preclude an unsafe turbine overspeed. The response of the controllers takes into account the residual steam in the piping between the valves and the turbine.

The EHC includes turbine trip indication devices for an electric solenoid trip.

### **10.2.3.17      Hydraulic Power Unit**

The major components of the HPU include a fluid reservoir, two independent and parallel pumping systems, fluid coolers, bladder accumulators, electric space heater, air dryer-breather, and a built-in fluid transfer and filtering unit.

The drain for the HPU is manually connected to drums to drain the reservoir.

The HPU supplies HP hydraulic fluid directly to the control solenoid on the TSVs, TCVs, intermediate stop and intercept valves, and TBVs for actuator control, and to the trip devices in the TMA. The unit can accommodate both steady-state and transient requirements.

The fluid is supplied to all components at the correct temperature and required cleanliness. The unit is equipped with chemically active filters to maintain the properties of the fluid over long service times. The two completely redundant pumping systems allow online maintenance of the standby pumping system. The HPU has alarms and pressure transmitters which auto start the standby pumping system. The EHC trips the turbine should a malfunction occur where the EHC cannot maintain and/or develop adequate pressure.

The reservoir has sufficient capacity to hold all of the hydraulic fluid for the turbine hydraulic system.

The reservoir is fabricated out of stainless steel.

The reservoir includes a cover plate to provide access to the reservoir for cleaning and inspection.

The reservoir includes internal baffling to allow any entrapped air in the return fluid adequate time to detrain before passing into the pump suction line and to separate the hot drain fluid from the rest of the fluid.

The reservoir contains a float type level gauge and level transmitter that determines the correct reservoir fluid level. This level transmitter incorporates low-low/low/high level switches that alarm.

The reservoir includes a temperature element that alarms for high and low fluid temperature.

A pressure compensator on each pump maintains a preset pressure throughout the delivery range i.e., as system demand changes, the pump automatically adjusts its stroke to meet the flow demand while maintaining system pressure.

If the pump dead heads, the pump is automatically shut off.

A wire mesh suction strainer is installed in the pump intake. The suction strainer provides a visual condition indicator during operation, showing suction status.

A full-flow HP filter is installed on the outlet of the pump. The full-flow HP filter includes an electric differential pressure switch that will alarm when the differential pressure has reached the maximum set point, indicating that the element requires changing.

Pumping discharge tubing includes relief valves that discharge the pump output back into the reservoir if pump pressure exceeds the setting of the valves.

Pump discharge tubing includes an automatic air bleed valve that discharges into the reservoir.

Two 100% capacity fluid-to-water shell and tube heat exchangers are provided to cool the fluid returning to the reservoir from the valve actuators and trip devices. One fluid cooler is used while the other is on standby.

Bladder type accumulators are connected to the hydraulic header in the supply manifold to provide an immediately available reserve of fluid for transient flow requirements.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Isolation valves in the tubing permit accumulators to be serviced while the HPU is in operation. The valves also permit checking of the pre-charge pressure by isolating an accumulator group and bleeding its fluid contents to drain.

Two redundant electric space heater units are included on the HPU.

The space heater units keep the fluid in the reservoir, accumulators, and tubing of the standby system warm during cold weather operation.

The space heaters are operated automatically by a thermostatic control in the HPU. The control senses temperature in the fluid reservoir and keeps the heaters on whenever the fluid temperature is below the low temperature set point.

An air dryer removes moisture from the air breathed by the fluid reservoir during HPU operation.

A built-in, dual operating-mode transfer and Fuller's earth filtration unit consists of a motor-driven transfer pump, relief valves, a filling and discharge hose, Fuller's earth filters, and a backup filter.

The reservoir is filled and drained by the transfer and Fuller's earth filtration unit.

The transfer filtering operation mode is used to filter the hydraulic fluid to a low particle size and contaminant level while the reservoir is being filled or drained. For this operation, fluid is passed through the Fuller's earth filters and the backup filter to the reservoir.

The Fuller's earth filtering operation mode maintains the fluid at a low neutralization number by absorbing acids and water and maintaining the chlorine content of the fluid within the recommended specifications. During the filtering operation, fluid from the reservoir flows through the Fuller's earth cartridges in filter housings through the backup filter and then back to the reservoir. The Fuller's earth filtering system is in operation during the Turbine Hydraulics system flush, and then kept in continuous operation when the turbine is in-service. For extended maintenance outages, the Fuller's earth filtering system is operated to maintain fluid operating conditions.

#### **10.2.3.18 Trip Manifold Assembly**

The TMA is designed as the final control element of the TGCS emergency overspeed protection system. The TMA consists of a duplex (parallel) arrangement of two triple redundant Electronic Trip Device solenoid valve circuits, Circuits 'A' and 'B', configured as a single integrated hydraulic circuit. The design is optimized for normal turbine generator operation with both circuits in-service but is capable of operating with one circuit isolated for online maintenance. Refer to Chapter 7 for information pertaining to triple redundancy.

Each TMA circuit includes three Electronic Trip Device valve assemblies, which are arranged hydraulically for two-out-of-three voting. The three pairs of isolation valves per circuit are arranged in a normally closed two-out-of-three voting circuit separating the Fluid Actuator Supply fluid supply from the Emergency Trip System header. Each two-out-of-three hydraulic circuit arrangement results in three parallel flow paths, each path passing through two valves in series. Trip and reset valve position transducers are provided for indication, fault annunciation, and test permissives for online test sequencing.

The TMA has a continuous-flow transfer valve assembly allowing for pressure isolation of either Electronic Trip Device circuit to permit online service.

The Electronic Trip Device solenoid valves direct Emergency Trip System fluid pilot pressure to the air relay dump valves, dump valves, and shut-off valves.

### **10.2.3.19 Electro-Hydraulic Control Valves and Tubing**

Electronic Trip Device solenoid valves are arranged for two-out-of-three voting logic in the TMA. The Electronic Trip Device solenoids are controlled by TGCS and by the diverse overspeed trip system. When the TMA is in normal operation with the Electronic Trip Devices energized, the Emergency Trip System fluid flows to pilot the air relay dump valves, flow through the fast-acting solenoid valves, hold the dump valves closed with the Emergency Trip System fluid pressure, and flow to pilot the shut-off valves to allow fluid actuator supply fluid to flow either to the applicable solenoid operated test valves or the servo valves.

The fast-acting solenoid valves, solenoid operated test valves, servo valves, dump valves, and shut-off valves are located with their associated actuators on the TSV, TCV, intermediate stop and intercept valves, and TBV.

Emergency trip system header dump valves are pilot operated with customized spools to maintain two separate and isolated flow paths. The dump valves are held closed by the Emergency Trip System fluid pressure. The dump valves have a spring bias to rapid close the TSVs, TCVs, intermediate stop and intercept valves with the release of the Emergency Trip System fluid pressure.

Air relay dump valves are Emergency Trip System fluid pilot-operated valves and used to establish the Emergency Trip Air System pressure in order to operate the extraction system non-return check valves. Actuation of the air relay dump valves causes the non-return valves air cylinders to be vented, allowing the non-return valves to close thereby preventing turbine overspeed.

Fluid actuator supply header isolation valves are two-way pilot-operated unbalanced logic cartridge valves arranged in pairs to simulate a single four-way valve. Each valve pair is piloted by a single Electronic Trip Device.

Emergency Trip System header shut-off valves are Emergency Trip System fluid pilot-operated with separate and isolated flow paths to admit the applicable fluid actuator supply fluid supply to the solenoid operated test valves or the servo valves for TSV, TCV, and intermediate stop and intercept valve actuation.

The transfer valve assembly consists of continuous-flow inlet and discharge valves that are used to hydraulically isolate either hydraulic Circuit A or B during normal operation without interruption of the fluid supply to the Emergency Trip System header.

Turbine hydraulic system tubing is designed to the requirements of ASME B31.1 (Reference 10.2-3).

The design of the hydraulic lines is fail-safe, meaning that loss of hydraulic pressure results in a turbine trip. All hydraulic tubing is constructed of stainless steel. HP hydraulic tubing is installed with welded connections where possible to reduce the potential for leaks. The tubing is supported in such a manner as to prevent vibration yet provide adequate flexibility to permit movement of end connections to valves. Where tubing passes through clearance holes in the floor or walls, the tubing is located such that the maximum tubing motion due to expansion does not result in interference between the tubing and the side of the clearance hole. All vent lines are routed such that air flow is not blocked by fluid pockets.

## **10.2.4 Turbine Integrity**

### **10.2.4.1 Materials Selection**

Turbine rotor integrity is provided through the combined use of selected materials with suitable toughness, analyses, testing, inspections, and operating procedures. Turbine components and

valves facilitate in-service testing and inspection (including all high stress regions) at intervals in accordance with industry practice (i.e., Boiling Water Reactor Owner's Group (BWROG) guidance) or as required to meet turbine missile generation probability requirements.

#### **10.2.4.2 Turbine Missile Probability Analysis**

Refer to Chapter 15, Subsection 15.6.1.2 which presents the methodology for performing a qualitative assessment for determining the probability of turbine generated missiles and associated results.

#### **10.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.2-2 for system interfaces.

#### **10.2.6 System and Equipment Operation**

##### **10.2.6.1 Normal Operation**

During normal operation, the MS TSVs and intermediate stop and intercept valves are fully open, with control valves modulating as required to maintain reactor pressure. Specifically, in normal operation, the RPC sends a flow demand signal to the turbine generator controller that changes a flow demand signal into a turbine control valve position demand signal. Because of the BWR's strong negative void coefficient, the control scheme in normal power operation cannot be "boiler follows turbine," since opening the control valves would lower reactor pressure and increase voiding which lowers reactor power and opposing the desired change. Instead, the design is "turbine follows boiler." Reactor power is first changed, and the turbine control valves open or close, appropriately increasing or decreasing load.

Operation of the turbine generator is controlled by the TGCS.

The reactor pressure controller controls the turbine control valves through the TGCS to regulate reactor pressure. The normal function of the TGCS is to generate the position demand signals for the main stop valves, main control valves, and the intermediate stop and intercept valves.

The Primary AC Bearing Oil Pump automatically supplies the turbine generator bearings and turning gear, if needed, with lube oil at the required pressure and temperature. If the Primary AC Bearing Oil Pump should fail, then the Secondary AC Bearing Oil Pump automatically supplies the system with lube oil.

During all operating modes, the lube oil is purified and conditioned by the integrally mounted lube oil conditioner. Lube oil is continuously circulated to the reservoir by a jockey pump that is located on the lube oil conditioner and drained back to the reservoir by gravity.

During Extraction Steam System normal operation, heating steam is transported to the FW heaters, with the exception of the No. 6 heater, and the MSR second stage reheaters. Both the motor-operated isolation valves and air-assisted non-return valves are open. Drain line valves are open and close as a function of the level of condensed steam in the drain pot. Extraction Steam System normal operation also includes isolating heaters:

1. LP FW Heaters Isolated: The LP FW heater is isolated when a high-high level setpoint is detected by the level transmitters in any of the three stages of LP FW heaters. This isolation protects the turbine from water induction. High-high level FW isolation results in the following actions:
  - a. The air-assisted non-return valve is released to close in the extraction line of the affected FW Heater No. 3
  - b. The remaining first and second stage extraction lines remain in normal operation

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

- c. The drain valve in the drain line upstream of the air-assisted non-return valve are signaled open
2. HP FW Heaters isolated: HP FW Heaters No. 4 and No. 5 are individually isolated when a high-high level setpoint is detected by their associated level transmitters. This trip protects the turbine from water induction. High-high level FW heater isolation results in the following actions:
  - a. The extraction isolation valves close, and the air-assisted non-return valve are released to close in the extraction lines to the affected FW Heaters No. 4 and No. 5. The sixth stage FW heater does not contain an air-operated non-return valve.
  - b. The drain valves in the drain lines upstream of the extraction isolation valves are signaled open

During normal operation, one pump from the EHC Pump Unit is sufficient to supply the Fluid Actuator Supply fluid for the HP hydraulic system. The pumps have a capacity of 100% duty each. The standby pump provides a reliable backup should the operating pump fail to provide the necessary pressure. Pressurized fluid from the HPU is pumped to the servo valves on the TCVs, intercept valves, and TBVs; the solenoid operated test valves on the TSVs and intermediate stop valves; and to the EHC.

During EHC normal operation, all Electronic Trip Devices are energized open, all shut-off valves are open, and all dump valves are held closed by the Emergency Trip System fluid, enabling the pressurization of the turbine EHC header. Also, during EHC normal operation, the air relay dump valves are reset to establish the Emergency Trip Air System pressure, which assists the extraction system non-return check valves in fast closure. The EHC system is able to support testing for the fast closure of the TSVs, TCVs, and the intermediate stop and intercept valves and fast opening of the TBVs. The EHC system has the ability to test the TMA when the turbine is online.

#### **10.2.6.2 Normal and Emergency Shutdowns**

In the event of an unsafe operating condition, the TGCS controls initiate trip signals to the turbine. The TGCS provides automatic and manual controls to unload the unit during controlled shutdowns.

The main turbine stop and control valves, and intermediate stop and intercept valves close upon actuation of the Emergency Trip System within a time period to preclude unsafe turbine overspeed.

In the event of a turbine trip with the closure of TSVs and TCVs, TASS provides sealing steam to the turbine until main condenser vacuum break. However, in the event of a turbine trip with a closure of the Main Steam Reactor Isolation Valves and/or Main Steam Containment Isolation Valve (MSCIV), the main condenser vacuum will break on loss of offgas steam, but the condenser vacuum pumps will be automatically started.

During a plant shutdown, the Primary or Secondary AC Bearing Oil Pump and the Primary or Secondary Lift Pump supply lube oil to the turbine generator bearings, couplings, and turning gear until the turbine speed is zero. The Primary or Secondary Lift Pump automatically start when the main turbine rotor speed decreases to near the point of an oil wedge breakdown.

During shutdown, the Extraction Steam System motor-operated isolation valves and air-assisted non-return valves are closed. Extraction line drain valves are open during shutdown. The shutdown of the system is carried out with the shutdown of the plant. The condensate produced during shutdown is drained to the condenser.

A turbine trip signal automatically isolates the shell side of FW Heaters No. 4 and No. 5 by closing the motor-operated isolation valves. The drain line valves are open.

The turning gear also operates during shutdown as long as the turbine is not under maintenance. During turning gear operation, the Primary Lift Pump supplies lube oil to the turbine generator rotor to reduce the turning gear torque requirements when the turbine is not in operation. If the Primary Lift Pump should fail, then the Secondary Lift Pump supplies oil to the bearings.

During a turbine trip, the HPU continues to supply Fluid Actuator Supply fluid to the servo valves on the TBVs to keep them open to discharge steam to the main condenser.

During shutdown, the HPU continues to supply hydraulic fluid to the TSVs, TCVs, intermediate stop and intercept valves and TBVs for actuation.

### **10.2.6.3 Off-Normal Operational Concept**

#### **10.2.6.3.1 Turbine Overspeed Protection System**

The following component redundancies are employed to provide turbine overspeed protection:

- Main stop valves/control valves
- Intermediate Stop and Intercept Valves
- Normal speed control/Primary overspeed trip/Emergency overspeed trip
- Fast-acting solenoid valves/emergency trip fluid system

In addition to the normal speed control function provided by the TGCS, a separate and redundant turbine overspeed protection system is included to minimize the possibility of turbine rotor failure and turbine missile damage. Additionally, the valve arrangements and closure times are such that a failure of any single valve to close will not result in unsafe turbine overspeed in the event of a trip signal.

The normal speed control system comprises a first line of defense against turbine overspeed. This system includes the MS control valves, intermediate stop and intercept valves, and fast-acting valve-closing functions within the TGCS. An increase in speed above the setpoint results in the closure of the main turbine control valves and intermediate stop and intercept valves in proportion to the speed increase. Rapid turbine acceleration resulting from a sudden loss of load at higher power levels normally initiates the fast-acting solenoids via the normal speed control system. The fast-acting solenoids rapidly close the main turbine control and intermediate stop and intercept valves irrespective of the current turbine speed. The normal speed control system is designed to limit peak overspeed resulting from a loss of full load, to at least 1% below the overspeed trip set point. Typically, this peak speed is in a range of 106-109% of rated speed, with the overspeed trip set point at 110% of rated speed. All turbine steam control valves and intermediate stop and intercept valves are fully testable during normal operation. The fast-closing feature, provided by action of the fast-acting solenoids, is testable during normal operation.

If the normal speed control should fail, the overspeed trip devices close the steam admission valves including the main and intermediate stop valves. This turbine overspeed protection system comprises the second line of defense against turbine overspeed (redundant and highly reliable). The overspeed protection system is designed to ensure that failure of the normal speed control system does not result in turbine speed exceeding 120% of rated speed. In addition, the components and circuits comprising the turbine overspeed protection system are testable when the turbine is in operation.

The failure of any single component does not cause the rotor speed to exceed the emergency overspeed.



The overspeed sensing devices are located in the turbine front bearing standard and are therefore protected from the effects of missiles or pipe breakage.

The hydraulic units are fail-safe, any loss of hydraulic pressure results in a turbine trip.

The main stop valves and control valves provide full redundancy in that these valves are in series and have independent control signals and operating mechanisms. Closure of both or either stop valves or control valves effectively shuts off all MS flow to the HP turbine. The intermediate stop and intercept valves are also in series and have independent control signals and operating mechanisms. Closure of either valve or both valves in each of the two intermediate stop and intercept valves effectively shuts off steam flow to the two LP turbines. This arrangement is such that failure of a single valve to close does not result in a maximum speed exceeding 120% of rated speed. The turbine is designed for a 120% overspeed.

#### ***10.2.6.3.2 Turbine Lube Oil System Off-Normal Operation***

The DC Emergency Bearing Oil Pump starts automatically and provides the lube oil required by the turbine generator in the event that the Primary and Secondary AC Bearing Oil Pumps are inoperative or in a loss of AC power. A controlled shutdown is initiated in the event that the DC Emergency Bearing Oil Pump is the only operable pump supplying lube oil, as failure of this pump will result in loss of oil to the bearings during coast down.

### **10.2.7 Instrumentation and Control**

#### **10.2.7.1 Instrumentation**

This section describes the instrumentation associated with the MTE. Instrumentation is provided for monitoring of thermal and hydraulic parameters and initiating alarms and automatic shutdown of the turbine in the event of an unsafe condition.

##### ***10.2.7.1.1 Main Turbine Equipment***

The main turbine contains instrumentation to:

- Monitor and control system operations
- Alarm abnormal system operation

Turbine generator supervisory instrumentation is provided for operational analysis and malfunction diagnosis.

The TSV and TCV are provided with pressure transmitters on the outlet of the valves.

The turbine contains Turbine Supervisory Instrumentation which monitors the following parameters:

- Vibration and eccentricity
- Thrust bearing wear
- Exhaust hood temperature
- Bearing metal temperatures
- Shell temperature
- Valve positions
- Shell and rotor differential expansion
- Shaft speed, and control valve inlet pressure indication

- Steam chest pressure

#### **10.2.7.1.2 Turbine Gland Seal System**

TGSS pressure transmitters are provided for the following:

- GSC exhaust blower suction pressure
- Steam seal header pressure
- TASS supply pressure
- Seal Steam dump valves differential pressure

A pressure indicator and transmitter is provided for the GSC exhaust blower suction pressure. Temperature indicators and transmitters are provided for the following:

- Steam seal header temperature
- GSC exhaust blowers suction temperature

A level indicator and level transmitter is provided for the GSC.

#### **10.2.7.1.3 Turbine Lube Oil System**

The instrumentation features of the TLOS include application of pressure transmitters for the following:

- Primary AC Bearing Oil Pump discharge pressure
- Secondary AC Bearing Oil Pump discharge pressure
- DC Emergency Bearing Oil Pump discharge pressure
- Lift Oil Pump discharge pressure
- Lift Oil Pump suction pressure
- Lube Oil Transfer Pump discharge pressure
- Lube Oil Transfer Pump suction pressure
- Filter 'A' differential pressure
- Filter 'B' differential pressure
- Lube Oil Conditioner filter differential pressure
- Lube Oil bearing header supply pressure
- Lube Oil Reservoir vacuum monitoring

Temperature elements are provided for the following:

- Lube Oil Reservoir immersion heater control

Temperature elements and transmitters are provided to indicate the following:

- Cooler outlet oil temperature
- Lube Oil Reservoir temperature
- Main Turbine bearing drain oil temperature (typical for all bearings including the Thrust Bearing)

Level transmitters are provided for the following:

- Lube Oil Reservoir level

Level gauges and transmitters are provided for the following:

- Lube Oil Storage Tank clean compartment
- Lube Oil Storage Tank dirty compartment
- Lube Oil Reservoir
- Lube Oil Conditioner water drain level

Flow gauges and transmitters are provided for the following:

- Lube Oil cooler 'A' oil vent flow
- Lube Oil cooler 'B' oil vent flow
- Lube Oil filter 'A' vent flow
- Lube Oil filter 'B' vent flow
- Bearing drain oil flow
- Lube Oil Conditioner drain water flow

#### **10.2.7.1.4 Extraction Steam System**

Pressure transmitters are provided to monitor the following:

- Pressure upstream of the drain line for LP FW Heater No. 3
- Pressure upstream of the drain line for HP FW Heater No. 4
- Pressure upstream of the drain line for HP FW Heater No. 5
- Pressure at the inlet of LP FW Heater No. 3
- Pressure at the inlet of HP FW Heater No. 4
- Pressure at the inlets of HP FW Heater No. 5

Temperature elements and transmitters are provided to monitor the following:

- Temperature at the extraction inlet nozzles of LP FW Heater No. 3
- Temperature at the extraction inlet nozzles of HP FW Heater No. 4
- Temperature at the extraction inlet nozzles of HP FW Heater No. 5

Test points and transmitters are provided to test the following:

- Temperature outside the condenser, but upstream of the drain line for LP FW Heater No. 3
- Temperature upstream of the drain lines for HP FW Heater No. 4
- Temperature upstream of the drain line for HP FW Heater No. 5.

#### **10.2.7.1.5 Electro-Hydraulic Control**

Pressure transmitters are provided for the following:

- HPU pump discharge pressure

- Fuller's earth filter differential pressure
- Backup filter differential pressure
- TMA Circuit 'A' pressure
- TMA Circuit 'B' pressure
- HPU pump filter differential pressure
- Hydraulic header pressure

Temperature element, gauge and transmitters are provided for the HPU fluid reservoir temperature.

Level transmitters are provided for the following:

- HPU fluid reservoir low-low level
- HPU fluid reservoir low level
- HPU fluid reservoir high level

Level gauges, transmitters, and position elements are provided for the following:

- TMA Circuit 'A' reset position
- TMA Circuit 'B' reset position
- TMA Circuit 'A' trip position
- TMA Circuit 'B' trip position

#### **10.2.7.2 Controls**

##### **10.2.7.2.1 Main Turbine Equipment**

The following turbine system conditions are alarmed in the Main Control Room (MCR):

- Loss of speed signal
- High thrust bearing wear
- Excessive rotor/shell/differential expansion
- High exhaust hood temperature
- High condenser pressure
- Overspeed
- High vibration
- High bearing temperature
- Zero rotor speed
- High rotor eccentricity
- Power Load Unbalance

The turbine stop and control valves and the intermediate stop and intercept valves close to trip the turbine on the following signals:

- Emergency trip push buttons in control room

- High moisture separator level
- High condenser pressure
- Low lube oil pressure
- High LP turbine exhaust hood temperature
- High reactor water level
- Excessive thrust bearing wear
- Overspeed (Primary and Emergency trip systems)
- Manual trip from the front standard
- Loss of stator coolant
- Low hydraulic fluid pressure
- Any generator trip
- Loss of TGCS electrical power
- Excessive turbine shaft vibration
- Loss of two speed probes – both within the primary set, or within the emergency set
- Loss of two pressure control channels

#### **10.2.7.2.2 Turbine Gland Seal System**

TGSS MCR Alarms include the following:

- High GSC blower suction pressure
- Low steam seal header pressure
- High GSC discharge radiation
- High and low steam source temperature and pressure
- High and low GSC level

TGSS MCR displays include the following:

- Status of blowers
- Status of all steam feed valves
- Status of all steam feed isolation valves
- Steam seal header pressure and temperature
- Pressure and temperature of steam sources
- GSC blowers suction pressure

The following paragraphs describe key control features of the turbine system.

The turning gear is provided with manual and automatic controls. When in automatic control, the turning gear engages and starts upon zero turbine rotor speed and sufficient lube oil pressure and disengages at startup.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

During low load operations, the air-operated pressure control steam seal feed valve provides steam seal header pressure control using steam taken from the TASS. The TASS comes off of the MSL equalizing header which is upstream of the TSV/TCV. This valve fails open on loss of signal or air. The steam seal feed valve seals the turbine by feeding steam up to 50% rated throttle pressure operating conditions. The steam seal feed valve can also operate with 100% rated throttle pressure during a full-load rejection. The steam seal feed valve has the ability to reduce the TASS source pressure by throttling.

At light loads and startup, the seal steam dump valve is closed, and the steam seal feed valve regulates the steam seal header pressure. If required during very low steam supply pressure conditions or with worn packing clearances, the bypass is used in parallel with steam seal feed valve when operating conditions are below 50% rated throttle pressure. When TASS pressure is high enough to maintain the steam seals automatically, the bypass is closed.

A manually controlled isolation valve is located upstream of steam seal feed valve.

TASS may be used at all loads as a sealing steam source if required. Steam is supplied by the air-operated pressure control Steam Seal Auxiliary Feed Valve. A manually controlled, air-operated valve is located downstream for auxiliary steam isolation.

The valves are arranged for fail-safe operation to protect the turbine. The feed valves fail open on the loss of either signal or supply air, and the seal steam dump valve closes with diaphragm failure. If, for any reason the automatic valves are out of order, control of the steam seal header pressure is achieved by operation of the bypass.

One GSC blower is operating whenever the TGSS is in-service with the other in standby. The standby blower automatically starts on trip of the lead blower.

The blower automatically stops on detection of blower discharge high radiation level.

#### **10.2.7.2.3 Turbine Lube Oil System**

TLOS provides a signal to control cooling water through the Lube Oil Coolers based on lube oil outlet temperature.

Primary and Secondary AC Bearing Oil Pumps automatic pump control:

1. The Primary and Secondary AC Bearing Oil Pumps are arranged in parallel with only one pump operating at a time.
2. Should the operating pump fail, the standby pump automatically starts due to low discharge oil pressure.

DC Emergency Bearing Oil Pump automatic pump control:

1. Should both the Primary and Secondary AC Bearing Oil Pumps fail, the DC Emergency Bearing Oil Pump automatically starts upon low oil pressure.
2. The DC Emergency Bearing Oil Pump automatically starts upon loss of all AC power.

TLOS design includes the ability to test the LP standby start functions of the Primary and Secondary AC Bearing Oil Pumps, and DC Emergency Bearing Oil Pump. The design also includes the ability to test the trip function of the Lift Pumps.

The following primary TLOS displays include the following:

- Primary AC Bearing Oil Pump status
- Secondary AC Bearing Oil Pump status

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

- DC Emergency Bearing Oil Pump status
- Primary Lift Oil Pump status
- Secondary Lift Oil Pump status
- Lube Oil Transfer Pump status
- Main bearing oil header pressure
- Lube Oil Cooler oil discharge temperature
- Vapor Extractor status

MCR Alarms include the following:

- Low bearing header oil pressure
- Low-low bearing oil pressure (turbine trip)
- Low Lube Oil Reservoir temperature
- High Lube Oil Reservoir temperature
- Low Lube Oil Reservoir level
- Low-low Lube Oil Reservoir level
- High Lube Oil Reservoir level
- High full-flow duplex filter A/B differential pressure
- Low Primary or Secondary Lift Pump discharge pressure
- High bearing oil drain temperature
- Lube Oil Cooler high discharge oil temperature
- Any Lube Oil pump auto start
- Trip of any TLOS pump

#### **10.2.7.2.4 Extraction Steam System**

The Extraction Steam System is operated and monitored from the MCR. Extraction Steam System controls and interlocks for the main components are described below:

1. The LP extraction line motor-operated block valve located in FW Heater No. 3 extraction line is used for Turbine Water Induction Protection. It is normally open and controlled from the MCR. The LP extraction motor-operated block valve is automatically closed if the turbine trips or upon receipt of an isolation signal from any level transmitter associated with the LP FW heater. The valve is fails-as-is.
2. The HP extraction line motor-operated block valve is used for Turbine Water Induction Protection. It is normally open and controlled from the MCR. The HP extraction motor-operated block valve is automatically closed if the turbine trips, upon receipt of an isolation signal from the level transmitters associated with the affected HP FW heater, or the condensate isolation valves to the FW heater are not 100% open. The valve is fails-as-is.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

3. Each air-operated level-controlled drain-pipe block valve is normally closed. They open automatically if the extraction isolation valve is not open, upon a high-high drain level signal, upon a turbine trip, or a low turbine load. The valves fail open.
4. The air-assisted non-return check valves are used to prevent turbine overspeed. The valves are controlled by the TGCS and when a turbine trip is detected, the valve closes. The air-assisted non-return check valves are released to close when a high-high level signal is received. These valves fail close.

Extraction Steam System displays, and alarms include, but are not limited to:

1. MCR displays
  - a. Measurements from all instruments described in Subsection 10.4.7.2
  - b. Motor-operated valve position
  - c. Air-assisted check valve position
  - d. Air-operated drain valve position
  - e. FW heaters turbine water induction prevention valves in test
2. MCR Alarms
  - a. LP in the extraction line
  - b. Extraction isolation valves not 100% open
  - c. High level in the extraction drain level line

#### **10.2.7.2.5 Electro-Hydraulic Control**

The EHC system provides a signal to maintain water flow through the HPU coolers based on hydraulic fluid outlet temperature.

HPU Main Pumps automatic pump control:

1. The HPU Main Pumps are arranged in parallel with only one pump operating at a time.
2. If the operating pump fails, a drop in hydraulic fluid pressure provides a signal to automatically start the standby pump.

When the TMA system receives a signal from TGCS on a loss of pressurized fluid, the turbine hydraulic system:

1. Trips the turbine by releasing the Emergency Trip System fluid pressure below each Dump Valve at the TSV, TCV, and Intermediate Stop and Intercept Valve control solenoid
2. Trips the pilot-operated air relay dump valves to allow for fast closure of the extraction system non- return valves

The MCR displays include, but are not limited to, the following:

1. HPU Main Pumps operating parameters, including discharge pressure, operating mode, and status
2. HPU Main Pumps motor operating parameters, including current and electrical equipment status
3. Hydraulic header pressure
4. Hydraulic fluid reservoir temperature



The MCR Alarms include, but are not limited to, the following:

- Low hydraulic fluid header pressure
- Low HPU fluid reservoir temperature
- High HPU fluid reservoir temperature
- Low HPU fluid reservoir level
- Low-low HPU fluid reservoir level
- High HPU fluid reservoir level
- High HPU full-flow filter differential pressure

### **10.2.8 Monitoring, Inspection, Testing, and Maintenance**

Turbine Water Induction Prevention valve testing for feedwater heaters is performed as required.

Component level inspections consist of visual, surface, and volumetric examination as discussed below:

- Visual, magnetic particle, and ultrasonic examination of all accessible surfaces of rotors
- Visual and magnetic particle or liquid penetrant examination of all turbine blades
- Visual and magnetic particle examination of couplings and coupling bolts

The ISI of valves important for avoiding overspeed includes the following:

1. The MS TSVs and control valves, and intermediate stop and intercept valves are tested under load. Test controls installed on the MCR panel permit full stroking of the stop valves, control valves and intermediate stop and intercept valves. Valve position indication is provided on the panel. Testing considers all resulting impacts, including level disturbances in the FW heaters due to fluctuating extraction pressures.
2. MS turbine stop and control valves, and intermediate stop and intercept valves are exercised at least once within each calendar quarter (or as required by the turbine missile probability analysis) by closing each valve and observing the remote valve position indicator for fully closed position status. This test also verifies operation of the fast-close function of each MS turbine stop and control valve and intermediate stop and intercept valves during the last few percent of valve stem travel. Fast closure of the intermediate stop and intercept valves is tested in a similar way if they are required to have a fast-close function different from the test exercise. This testing frequency complies with the BWROG turbine surveillance test program.
3. Tightness tests of the MS turbine stop and control valves are performed at least once per maintenance cycle by checking the coast down characteristics of the turbine from no load with each set of two valves closed alternately or using warm-up steam as an indicator with the valves closed.
4. All MS TSVs, control valves, and intermediate stop and intercept valves are disassembled and visually inspected once during the first two refueling or extended maintenance shutdowns. Subsequent inspections are scheduled in accordance with the BWROG turbine surveillance test program. The inspections are conducted for:
  - Visual and surface inspections of valve seats, disks and stems including:
    - Wear of linkages and stem packings

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

- Erosion of valve seats and stems
- Deposits on stems and other valve parts which could interfere with valve operation
- Distortions, misalignment, or cracks
- Inspection and cleaning of valve bushings
- Confirmation of bore diameters for proper clearance

Routine testing of the TLOS is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. Routine testing includes tests of the AC pumps and Emergency DC pump.

Routine testing of the EHC is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. The EHC is tested while the turbine is online to verify the system functions properly.

### **10.2.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA) in operational states.

Chapter 12, Section 12.2 describes the radioactive source terms in the BWRX-300 TB. The main potential source of airborne radioactivity within the TB is leakage from valves on large lines carrying HP steam. Noble gas airborne concentrations are expected to be negligible throughout the TB except for inside the SJAE cubicles. These areas are not normally occupied during operation, and the exhaust from these cubicles is exhausted to the environment after filtration to eliminate the possibility of contamination of adjoining areas.

As stated in Chapter 11, Subsection 11.5.5.7 the Process Radiation Monitoring Subsystem (PRM) provides radiation monitoring for the Main Turbine Gland Seal Steam Condenser Exhaust flow path.

The PRM and Area Radiation Monitoring Subsystem provide general monitoring located in the TB as discussed in Chapter 11, Subsection 11.5.5.

### **10.2.10 Performance and Safety Evaluation**

The turbine generator is not needed to effect or support a safe shutdown of the reactor. The turbine is designed, constructed, and inspected to minimize the possibility of any major component failure. The turbine has a redundant, diverse, and testable overspeed trip system to minimize the possibility of a turbine overspeed event.

Instruments, controls, and protective devices are provided to confirm reliable and safe operation. Redundant, fast actuating controls are installed to prevent damage resulting from overspeed and/or full-load rejection. The control system initiates a turbine trip upon reactor trip.

The turbine generator equipment shielding requirements and the methods of access control for required areas of the TB ensure that the dose limits for operating personnel are not exceeded. All areas in proximity to TG equipment are zoned according to expected occupancy times and radiation levels anticipated under normal operating conditions.

The MTE's physical layout provides protection to essential systems and components, as required, from the effects of high and moderate energy turbine generator system piping failures or failure of the connections(s) from the LP turbine exhaust hoods to the condenser due to missile

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

protection and overspeed. Failure of turbine generator equipment does not preclude safe shutdown of the reactor systems.

**10.2.11 References**

- 10.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.2-2 USNRC Regulatory Guide 1.115, "Protection Against Turbine Missiles."
- 10.2-3 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 10.2-4 EPRI-NP-5479, "Application Guide for Check Valves in Nuclear Power Plants," Electric Power Research Institute.
  
- 10.2-5 10 CFR 50 Appendix A(4), "Environmental and Dynamic Effects Design Bases."
- 10.2-6 ASTM TDP-2, "Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants," ASTM International.

**Table 10.2-1: Summary of Design Features and Performance  
Characteristics of the Power Conversion System**

Gross generator output, MW	302.8
Throttle steam pressure, MPa abs	Approximately 6.89
Throttle steam temperature, °C	Approximately 285
Reheat steam temperature, °C	Approximately 259
LP turbines exhaust pressure, mm HgA	Approximately 45
Extraction FW heaters in-service	5
Final FW temperature, °C	Approximately 241.3
Number of condensers	2
Type	Surface Condenser
Heat load	Approximately 567.2 MWt

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.2-2: Main Turbine Equipment Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System (NBS)	NBS provides steam to MTE	Turbine Stop Valve
Non-Safety Instrumentation and Controls	RPC (Steam Bypass and Pressure Control) and Turbine Controls System	Control system for MTE valves and pumps
Process and Radiation Monitoring System	Radiation surveillance of main turbine generator and its auxiliary equipment in the TB	Operational sampling point
Control Panel System	Monitor all equipment	Turbine and auxiliaries
Offgas System	Air and steam discharge from the GSC's blowers	Flange connection at the GSC exhaust fans
Condensate and Feedwater Heating System	Extraction steam and LP exhaust feed into Condensate and FW	HP Extraction LP Extraction
Moisture Separator Reheater	HP turbine extraction and exhaust goes into MSR to be reheated and then feed into the LP turbine intake	HP Extraction HP Exhaust LP intake
Turbine Bypass System	MTE provides hydraulic pressure to actuate TBVs	TBV hydraulic actuator
Generator	Generator Rotor connected to turbine, to generate power	Generator Rotor coupling to turbine
Main Condenser and Auxiliaries	LP turbine discharges into condenser. MTE provides steam of SJAE. Provides steam for steam seals	LP turbine exhaust flange and the condenser inlet flange

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Plant Cooling Water System	Provides cooling water to Lube oil and EHC coolers/heat exchangers	Inlet and outlet nozzle connections of Lube oil and EHC coolers/heat exchangers
Plant Pneumatics Equipment	Air supply for Air Operated Valves (AOVs)	Connection at the AOV port
Non-Safety Electrical Distribution system	Power for solenoid valves and pumps, uninterruptible AC power, uninterruptible DC power	Terminal connections at the electrical component for power
Equipment and Floor Drain System	Incidental waste from oil systems	Equipment drain tub
Turbine Building Structure	Accommodate loads for the MTE System	Turbine anchors with the TB Structure pedestal
Water, Gas, and Chemicals Pads	Provides water through a hose for GSC drain loop seal	Temporary connections

NEDO-33960 REVISION 0  
NON PROPRIETARY INFORMATION

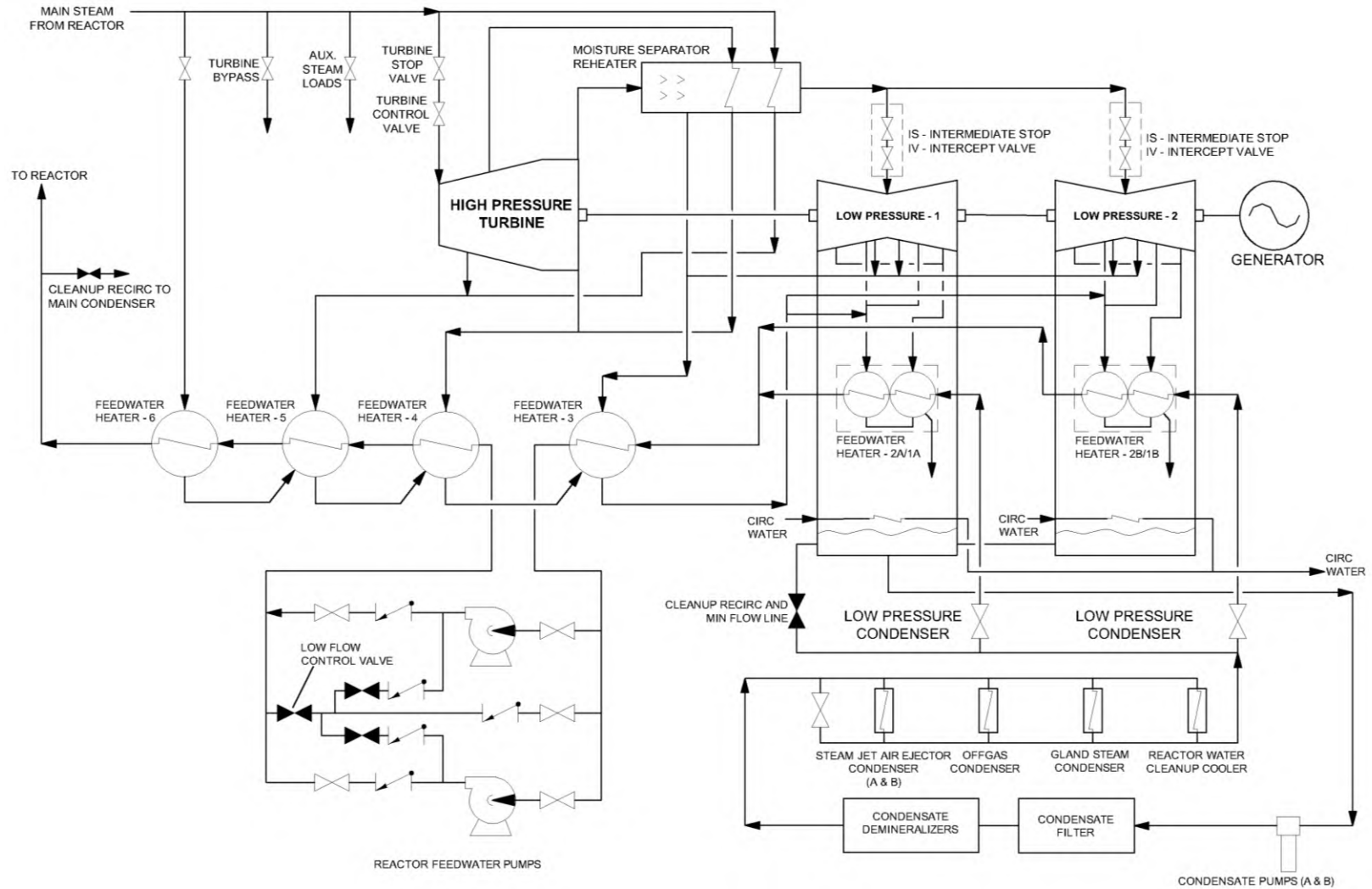


Figure 10.2-1: Power Cycle Schematic

### **10.3 Feedwater Systems**

#### **10.3.1 Condensate Filters and Demineralizer System**

The Condensate Filters and Demineralizers System (CFD) purifies the condensate to maintain reactor FW purity. The CFD uses filtration to remove suspended solids, including corrosion products, and ion exchange resin to remove dissolved solids from condenser leakage and other impurities. The CFD is a full-flow system that consists of high efficiency backwash type filters followed by mixed bed demineralizers.

##### **10.3.1.1 System and Equipment Functions**

###### **10.3.1.1.1 Normal Functions (Non-Safety Category)**

The CFD continuously operates during all modes of normal power plant operation, including startup and shutdown, receiving untreated condensate from, and supplying treated condensate to, the CFS. The CFD carries out the following functions:

1. Continuously removes dissolved and suspended solids and corrosion products from the condensate and from drains returned to the condenser hotwell, to limit accumulation of corrosion products. The series of filters and demineralizers through which FW is processed target the achievement of water chemistry requirements specified in Reference 10.3.1-2, Electric Power Research Institute, "BWRVIP-190 Revision 1: BWR Vessel and Internals Project," Volume 1, 2014, or most current version.
2. Removes impurities entering the power cycle due to condenser circulating water leaks as required to permit continued power operation within specified water quality limits as long as such condenser leaks are small. Continued operation with minor condenser circulating water leaks is controlled by site-specific chemistry threshold values.
3. Maintains the Condensate Storage Tank (CST) water quality as required for condensate makeup and miscellaneous condensate supporting services.
4. The CFD limits the entry of dissolved solids into the FW system in the event of large condenser leaks, such as a tube break, to permit a reasonable amount of time for orderly plant shutdown.
5. The CFD flow controls and bypass are arranged such that the condensate system flow is uninterrupted even in the presence of a single failure.
6. The CFD filters and removes ionic impurities from the Reactor Water Cleanup System (CUW) (Chapter 9A, Subsection 9A.2.2).

###### **10.3.1.1.2 Normal Functions (Safety Category)**

The system does not perform any Safety Category functions during normal conditions.

###### **10.3.1.1.3 Off-Normal Functions (Non-Safety Category)**

The CFD continuously operates during all modes of off-normal power plant operation, including turbine trip, low reactor load, and condenser leakage.

The CFD bypass can be used during startup and normal operations, in the event of an Anticipated Operational Occurrence resulting in a disruption of CFS flow through the CFD. The CFD bypass will remain open until the CFD flow is returned to normal, or the plant is brought to an orderly shutdown.

The CFD is not required to operate during or after a design basis event.



#### **10.3.1.1.4 Off-Normal Functions (Safety Category)**

The system does not perform any Safety Category functions during off-normal conditions.

#### **10.3.1.2 Safety Design Bases**

The CFD does not perform, ensure, or support any Safety Category function, and thus, has no safety design bases. The CFD is Non-Safety Class (SCN) and is categorized as Non-Nuclear Seismic.

The CFD is not required to operate during or after a design basis event. The design of the CFD meets CNSC requirements specified in REGDOC-2.5.2 Section 6.1 (Reference 10.3.1-1) as it relates to design for Defence-in-Depth and Section 7.1 as it relates to classification of SSC.

#### **10.3.1.3 Description**

Refer to Figure 10.3.1-1 which depicts the CFD.

CFD equipment, valves, and instrumentation are located within the Radwaste Building.

The CFD includes three (3) filter vessels, each rated at 50% of total FW flow with high efficiency backwash type filter elements arranged in parallel, complete with valves, piping, instrumentation, and filter backwash equipment. The condensate demineralizers include three (3) demineralizer vessels, each rated at 50% of total FW flow arranged in parallel, complete with valves, piping, controls, and instrumentation.

The CFD has connections to auxiliary skids and tanks to assist in tasks required to clean the condensate filters and replace spent resin in the condensate demineralizers. A fresh resin hopper is used to feed the demineralizer.

The CST transfer pumps will provide adequate pressure and condensate flow to transport resin to and from the demineralizers. The capability exists to wash resin injected into the condensate demineralizer prior to placing the demineralizer into standby mode.

The CFD is capable of handling the normal condensate flow while maintaining the required water quality.

Demineralizer influent and effluent conductivities are monitored by the Process Radiation and Environmental Monitoring System (PREMS) (Chapter 11, Section 11.5) to determine condenser leakage and resin exhaustion, respectively.

Total throughput capacity for each demineralizer and filter vessel is monitored. Flow rates of demineralizers and filters in-service are monitored and controlled.

Initial resin rinse for the mixed resins is performed in the offline condensate demineralizer tank. Rinse water is supplied from the Liquid Waste Management System (LWM) (Chapter 11, Section 11.2) CST and drained to the EFS (Chapter 9A, Subsection 9A.4.2).

Individual air-operated bypass valves are provided for both the condensate filters and condensate demineralizers. Each bypass valve is capable of handling full condensate flow.

Each bypass system includes a single automatic bypass valve with two manual isolating valves. These manual valves are used to isolate the automatic valve from the CFS.

Provisions are included to permit cleaning and replacement of the ion exchange resin. To perform the addition of new resins to the condensate demineralizers, one fresh resin hopper is included in the CFD design.

#### **10.3.1.3.1 Component Description**

The following paragraphs provide information that pertains to the major CFD components. Table 10.3.1-1 contains a summary of CFD components:

- **Condensate Filters**

The CFD includes backwash type high efficiency filters. The filters and vessels are designed to meet the requirements of ASME Boiler and Pressure Vessel Code (BPVC), Section VIII, Division 1 (Reference 10.3.1-3). The condensate filters receive condensate from the CFS and assist in purifying by filtering out suspended solids that have entered the condensate stream.

Filter vessels are constructed of corrosion resistant stainless steel or epoxy coated carbon steel.

- **Condensate Demineralizers**

Demineralizer vessels are constructed of corrosion resistant steel or lined with corrosion resistant material. The demineralizers are designed to meet the requirements of ASME BPVC, Section VIII, Division 1 (Reference 10.3.1-3). The mixed bed type condensate demineralizers receive condensate water from the CFS after being filtered by the condensate filters. The demineralizers assist in treating the condensate water by removing iron, copper, silica, and suspended solids.

- **Resin Traps**

The CFD includes resin trap housings located downstream of the condensate demineralizers. A resin trap is installed downstream of each demineralizer vessel to preclude gross resin leakage into the power cycle in case of vessel resin screen failure.

The resin traps are constructed of corrosion resistant stainless steel resin baskets.

- **Condensate Filter Bypass Valve**

A condensate filter bypass valve is provided for use during startup and in an off-normal condition until the condensate flow is returned to normal or the plant is brought to an orderly shutdown.

The condensate filter bypass valve is constructed of corrosion resistant stainless steel.

- **Condensate Demineralizer Bypass Valve**

A condensate demineralizer bypass valve is provided for use during startup and in an off-normal condition until the condensate flow is returned to normal or the plant is brought to an orderly shutdown.

The condensate demineralizer bypass valve is constructed of corrosion resistant stainless steel.

- **Air Surge Tank**

An air surge tank is provided in the CFD as a means of transporting spent resin from the condensate demineralizer drain to the Solid Waste Management System (SWM) (Chapter 11, Section 11.4) spent resin tanks. The air surge tank can also provide air for backwashing the filters when needed. The air surge tank receives air from the Plant Pneumatics System (Chapter 9A, Subsection 9A.4.1). The air surge tank is designed to meet the requirements of ASME BPVC, Section VIII, Division 1 (Reference 10.3.1-3).

- Fresh Resin Addition Hopper

The fresh resin addition hopper provides a location for storage of fresh resin before injection into a condensate demineralizer vessel.

The resin addition hopper is constructed of corrosion resistant stainless steel.

- Piping and Valves

The piping is designed to meet the requirements of ASME B31.1 for power piping (Reference 10.3.1-4) and B31.3 for process piping (Reference 10.3.1-5).

Isolation valves are provided on the inlet and outlet of each condensate filter vessel and each demineralizer vessel to allow the vessel to be out of service for backwashing, resin transfer or vessel maintenance.

Pressure taps between the automatic valve and the downstream isolation valve are used as test connections to indicate whether one of the valves is leaking.

#### **10.3.1.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically through material chemistry, heat treatment, contamination, and material processes controls. CFD components are constructed of stainless steel or lined with corrosion resistant materials as applicable.

#### **10.3.1.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.3.1-2 for system interfaces.

#### **10.3.1.6 System and Equipment Operation**

The CFD is capable of performing its system functions as described in Section 10.3.1 during the modes of operation described below.

##### ***10.3.1.6.1 Normal Operational Concept***

##### **Plant Startup Mode**

The CFD is used during plant startup to clean up the condensate, and to minimize the time required to achieve low power operation of the unit. The CFD provides high purity water to the CFS. The CFS provides recirculation flow paths to assist in cleaning the condenser hotwell and facilitate achieving the desired condensate water chemistry prior to sending FW to the reactor.

##### **Plant Normal Operation Mode**

The CFD operates continuously and provides the required reactor FW quality.

The CFD is designed to operate with two online demineralizers and two online filters. Demineralizers are sequenced as necessary to prevent resin channeling. The filters are sized to allow full-load operation with one filter out of service. Each filter is capable of handling approximately 50% of normal condensate flow.

During normal operation, the condensate pump discharge is routed through the CFD. The condensate enters the filters, which removes filterable solids. The effluent from the filters passes through mixed bed ion exchange demineralizers, which are also piped in parallel (one on standby) and where removal of dissolved impurities is accomplished. Effluent from the CFD is returned to the CFS.

During normal operation the CFD is capable of completing typical resin and filter maintenance activities, including backwashing, receiving fresh resin, resin transfer, discharging exhausted resin, placing a standby filter or demineralizer into service, and removing an operating filter or demineralizer from service.

### **CFD System Bypass**

The design includes bypass facilities capable of bypassing condensate flow around the CFD to provide continuous short-term condensate flow in the event of a condition within CFD which prevents treatment of the entire condensate flow. The CFD bypass can also be used during CFS startup of the condensate pumps.

Independent bypass valves are supplied for both the condensate filters and condensate demineralizers. The filter bypass valve automatically opens based upon differential pressure across the filters. The filter bypass valve is capable of passing full condensate flow. The demineralizer bypass valve opens automatically based upon the differential pressure across the demineralizers. The demineralizer bypass valve is capable of passing full condensate flow. The demineralizer bypass valve will open upon high temperature upstream of the CFD system to prevent damage to the resin media.

### **Plant Shutdown**

The CFD is not required to be in-service during unit shutdown.

#### ***10.3.1.6.2 Off-Normal Operational Concept***

### **Turbine Trip**

During and following a turbine trip, the CFD operates continuously and provides treated condensate to the CFS.

### **Reactor Low Load**

The CFD operates continuously and provides treated condensate to the CFS under low load conditions.

### **Condenser Tube Leak**

The CFD limits the entry of dissolved solids into the CFS and, in the event of a tube break, allows a reasonable amount of time for orderly plant shutdown.

#### **10.3.1.7 Instrumentation and Control**

The following defines some Instrumentation and Controls (I&Cs) required for system operation and control. Most operations are through the MCR and the following indications, alarms and controls are provided but are not limited to the items below.

Differential pressure transmitters are provided to monitor pressure drop across:

- The effluent and inlet filter vessel headers
- Individual filter vessels
- Effluent and inlet demineralizer vessel headers
- Individual demineralizer vessel
- Each demineralizer resin trap

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Flow transmitters are provided to monitor flow through:

- Each filter
- Each demineralizer vessel
- The demineralizer sluice water supply header
- The combined influent header of the condensate filters
- The combined influent header of the demineralizers

Temperature elements and transmitters are provided to monitor:

- The CFD inlet temperature
- The demineralizer sluice supply header temperature

Pressure transmitters are provided to monitor pressure of:

- The filter backwash water supply header
- Air supply header to the air surge tank
- Combined effluent header of demineralizers
- Service air supply header for resin transfer

The following defines the basic controls required for system operation and control.

The MCR is equipped with the appropriate instruments and controls to allow an operator the ability to perform the following operations as directed by procedure:

- Monitor filters and demineralizers in normal operation
- Control the filters and demineralizers in their various operating modes
- Remove a saturated filter from service
- Clean up the isolated filter by backwashing and place it back in operation
- Remove an exhausted demineralizer from service and replace it with a standby unit
- Transfer the resin inventory of the isolated demineralizer vessel into the spent resin tank for disposal
- Wash resin within an isolated demineralizer via slow water washing
- Transfer the fresh resin to any isolated demineralizer vessel

Interlocks are provided for the following:

- To prevent feed into a demineralizer with a high differential pressure across the demineralizer inlet isolation valve
- To prevent high temperature feed to demineralizers
- To open the bypass in case of high differential pressure across the condensate filters or demineralizers
- To prevent sending waste backwash water from the filter to the SWM if the SWM is unavailable

Conditions for the condensate filters are alarmed in the MCR:

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

- High influent conductivity (monitored by PRM)
- High differential pressure across condensate filter system
- High differential pressure across individual condensate filters

Conditions for the condensate demineralizers are alarmed in the MCR:

- CFD influent temperature high
- High differential pressure across condensate demineralizer system
- High conductivity at the outlet of each demineralizer vessel (monitored by PRM)
- High conductivity at the common effluent from the demineralizers (monitored by PRM)
- High differential pressure across each demineralizer
- High differential pressure across each demineralizer effluent resin trap

The following signals for the condensate filters are monitored in the MCR:

- Condensate filter status
- Condensate filters common inlet conductivity (monitored by PRM)
- Individual condensate filter flow and totalization
- Differential pressure across condensate filters
- Condensate filter combined effluent flow

The following signals for the demineralizers are monitored in the MCR.

- Condensate demineralizer status
- Condensate demineralizer flow and totalization
- Condensate demineralizer system differential pressure
- Condensate demineralizer vessel differential pressure
- Differential pressure across resin trap
- CFD inlet header temperature
- Condensate demineralizer effluent conductivity (monitored by PRM)
- Condensate demineralizers combined effluent conductivity (monitored by PRM)

#### **10.3.1.8 Monitoring, Inspection, Testing, and Maintenance**

Section 10.3.1.7 identifies some conductivity measurements that are monitored by PRM.

Pre-operational tests are performed on the CFD to ensure operability, reliability, and integrity of the system. Each filter and demineralizer vessel can be isolated during normal plant operation to permit testing and maintenance. Filters can be backwashed remotely.

Isolation valves are provided on the inlet and outlet of each condensate demineralizer vessel to allow the vessel to be out of service for backwashing, resin transfer, or vessel maintenance. CFD capacity is designed to allow full power operation with one condensate filter or demineralizer out of service for maintenance.

The equipment and components of the CFD are designed for ease of inspection and maintenance during plant operation. Routine testing of the CFD is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

The CFD design provides equipment removal paths and personnel access for maintenance. Shielding is provided to protect personnel from the concentration of radioactive material (activated corrosion products and fission products that are carried over from the reactor) in filters, demineralizers and other CFD equipment.

#### **10.3.1.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

Shielding is provided to protect personnel from the concentration of radioactive material (activated corrosion products and fission products that are carried over from the reactor) in filters, demineralizers and other CFD equipment.

#### **10.3.1.10 Performance and Safety Evaluation**

The CFD is capable of performing its system functions during the modes described in Subsection 10.3.1.6. The CFD does not perform any Safety Category functions during normal and off-normal conditions and therefore does not require a safety evaluation.

#### **10.3.1.11 References**

- 10.3.1-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.3.1-2 Electric Power Research Institute, "BWRVIP-190 Revision 1: BWR Vessel and Internals Project," Volume 1: BWR Water Chemistry Guidelines- Mandatory, Needed, and Good Practice Guidance.
- 10.3.1-3 ASME BPVC, Section VIII, Division 1, Rules for Construction of Pressure Vessels.
- 10.3.1-4 ASME B31.1, Power Piping, American Society of Mechanical Engineers, New York, NY.
- 10.3.1-5 ASME Code for Pressure Piping, B31.3, Process Piping.

**Table 10.3.1-1: Condensate Filter and Demineralizer System**

Parameter	Standard Plant Value*
<b>Condensate Filters:</b>	
Filter type	High efficiency
Number of vessels	3
<b>Condensate Demineralizers:</b>	
Demineralizer type	Deep bed polisher
Number of vessels	3
<b>Other System Features:</b>	
New resin addition hopper	1



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.3.1-2: Condensate Filter and Demineralizer System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Instrumentation and Controls	Non-Safety I&C provides Non-Safety I&C control. Valve logics, instrumentation (pressure, flow, temperature, level)	Non-Safety I&Cs Distributed Control and Information System (DCIS) termination cabinet
Process Radiation and Environmental Monitoring System	PREMS provides sample locations to test for process quality and conductivity	Isolation valves
Liquid Waste Management System	LWM provides the CST water for resin transfer, resin washing, and filter backwash processes	Isolation valve to the Sluice Water header
Solid Waste Management	SWM receives and stores condensate filter backwash waste in sludge tank, and demineralizer, spent resin waste in the spent resin tank	Backwash and Spent Resin discharge isolation valves
Condensate and Feedwater Heating System	CFS provides condensate water flow to CFD for filtration and demineralization. CFD provides clean effluent for FW heating process	Condensate Filters inlet header and outlet of the Condensate Demineralizers
Plant Pneumatics System	Provides air to pneumatically operated valves and compressed air for filter cleanup and resin transfer	Isolation valve to the air surge tank and valve actuators
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides SCN low voltage power	Electrical terminals of the power cables at the load end
Equipment and Floor Drain System	EFS provides location for drains and relief valve blowdowns	Interface is at the piping that is routed EFS

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

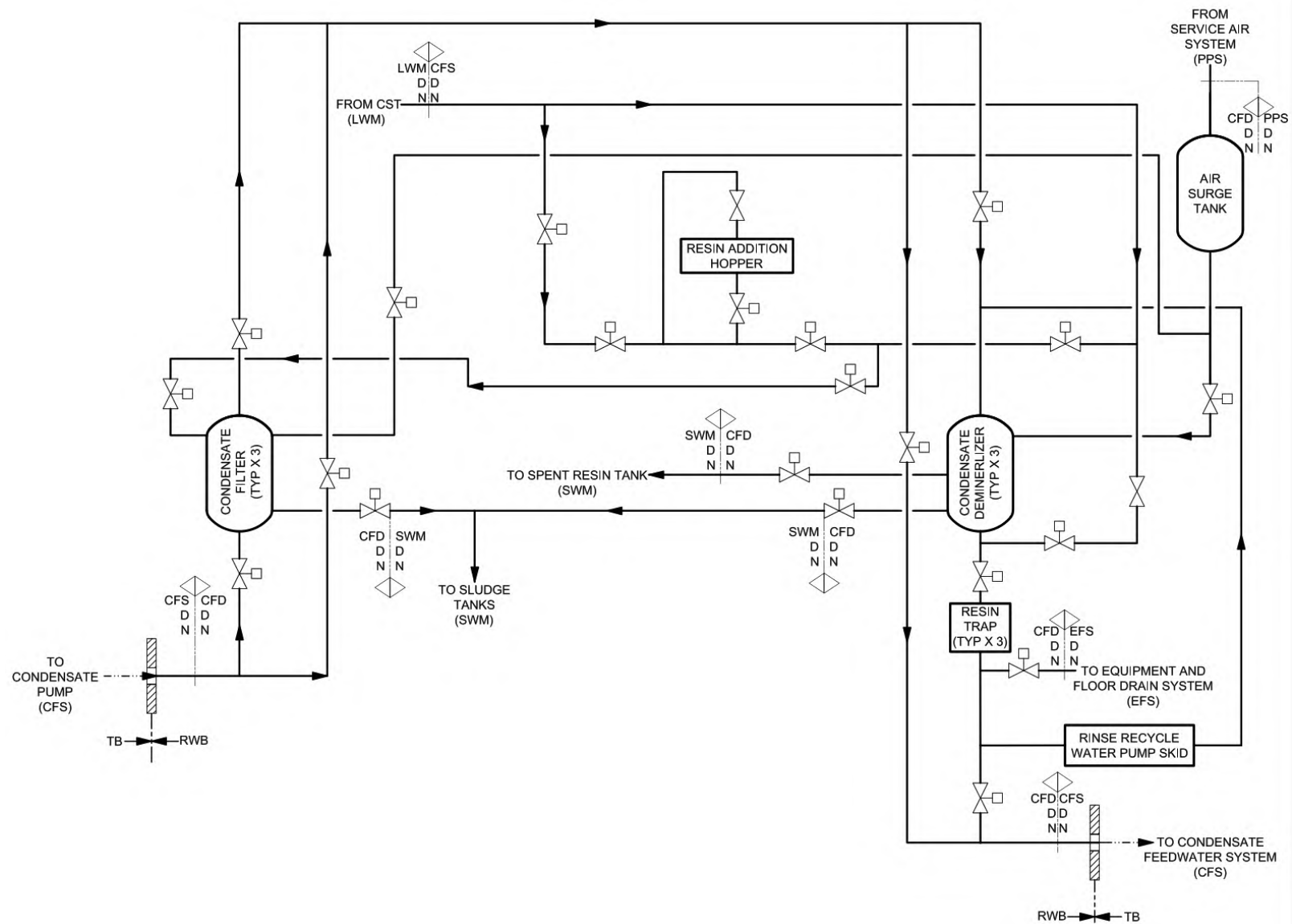


Figure 10.3.1-1: Condensate Filters and Demineralizer System

### **10.3.2 Condensate and Feedwater Heating System**

The condensate portion of the CFS is designed to pump condensate from the main condenser hotwell through the CFD and three stages of LP FW heating to the suction side of the reactor feed pumps. The FW portion of CFS is designed to pump FW through three stages of HP FW heating and deliver the FW to the reactor inlet nozzles.

#### **10.3.2.1 System and Equipment Functions**

##### **10.3.2.1.1 Normal Functions (Non-Safety Category)**

The CFS provides the following Non-Safety Category functions during normal conditions:

1. Provides vent paths for vapor and non-condensables in the closed FW heaters

##### **10.3.2.1.2 Normal Functions (Safety Category)**

The CFS provides the following Safety Category functions during normal conditions:

1. Supplies subcooled FW to the NBS to maintain the operating reactor water level band in the RPV whenever the condenser is available to reject decay heat.
2. Controls the steam demand to the 6th stage FW heater to automatically control the final FW temperature. This includes measuring final FW temperature.
3. Raises the temperature of the FW through FW heaters.
4. Provides normal and alternative drain paths for condensed steam in the FW heaters.
5. Supplies condensate to the Control Rod Drive (CRD) system.

##### **10.3.2.1.3 Off-Normal Functions (Non-Safety Category)**

The CFS does not provide any Non-Safety Category functions during off-normal conditions.

##### **10.3.2.1.4 Off-Normal Functions (Safety Category)**

The CFS provides the following Safety Category functions during off-normal conditions:

1. CFS provides the capability to measure FW pressure.
2. CFS provides the capability to modulate the flow of FW to the reactor for the purpose of reactor level control, based on signals from I&C systems.
3. Continues to supply condensate water following a single condensate pump trip.
4. Continues to supply FW following a single FW pump trip.
5. CFS provides the capability to monitor the operational status of the condensate and FW pumps.
6. CFS FW pump in standby will start when signaled to do so by I&C systems.
7. CFS provides the sensors to measure final FW temperature.
8. CFS provides the sensors to detect a FW line break outside containment.
9. CFS provides containment and system isolation of CFS piping penetrating the primary containment and will close when demanded from the SC1 I&C System or SC2 and SC3 I&C System.
10. CFS provides the capability to measure FW flowrate.

### **10.3.2.2 Safety Design Bases**

The CFS system is primarily SC3. The CFS provides for closure of the containment isolation valves (CIV)s and system isolation valves when signaled to do so as discussed in Section 10.3.2.1. The containment isolation components, system isolation components, leak detection components, and flow measuring devices are SC1. Containment isolation is provided in accordance with REGDOC-2.5.2 Section 8.6.6 (Reference 10.3.2-1).

Codes and standards related to pressure boundary and seismic classification are identified in the component descriptions in Subsection 10.3.2.3.1 as applicable.

The standby condensate and FW pumps will start when signaled to do so by the SC2 and SC3 Instrument and Control System. In support of this function, the CFS provides the capability to monitor the operational status of the FW pumps.

The CFS modulates FW flow to the vessel to control reactor vessel level.

The CFS provides for measurement of FW flow to provide input to the SC1 Instrument and Control System.

CFS measures final FW temperature and pressure and provides this information to the SC2 and SC3 Instrument and Control System.

CFS provides sensors to detect a FW line break outside containment.

CFS provides containment isolation of CFS piping penetrating the primary containment and will close when signaled to do so from the SC1 Instrument and Control System.

CFS provides system isolation of CFS piping penetrating the primary containment and will close when signaled to do so from the SC1 Instrument and Control System.

### **10.3.2.3 Description**

Figure 10.3.2-1 and Figure 10.3.2-2 depict the CFS flow diagrams. Figure 10.3.2-3 and Figure 10.3.2-4 depict CFS heaters and vents associated with the CFS.

The CFS is located in the TB, Reactor Building and Containment. During normal operation, the CFS provides a continuous supply of purified condensate water to the NBS (Chapter 5) at the required temperature, pressure, and flowrate.

The CFS consists of two (2) 100% capacity condensate pumps, three (3) LP FW heaters arranged series, one (1) Low Flow Control Valve, two (2) 100% capacity reactor FW pumps, three (3) HP FW heaters arranged in series, bypass valves around LP Heaters No. 1 and No. 2 and HP FW heater, piping, valves, and instrumentation.

A condensate pump discharge header connects the CFS to the CFD (Subsection 10.3.1). The CFS provides cooling to the CUW (Chapter 9A, Subsection 9A.2.2) heat exchanger, SJAЕ intercondensers, TGSS condenser, and offgas condenser. The CFS provides condensate water to the condenser hood spray, CRD pumps, and the CST for high level control in the condenser. A recirculation line to the main condenser is equipped with a control valve that connects to the auxiliary condensate loop downstream of the TGSS condenser is provided to ensure condensate minimum flow and auxiliary condenser cooling.

Downstream of the auxiliary condensate loop, FW heaters are provided for condensate and FW temperature control. Condensate pumps and FW pumps are provided to meet rated pressure and flowrate entering the NBS. The condensate pump minimum flow/short cycle cleanup line provides water inventory control during multiple operating modes, in addition to the minimum flow and cleanup functions.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

The shell side of the FW heaters drain in a cascading order from FW Heater No. 6 to FW Heater No. 1.

The final stage HP FW heater (No. 6) accommodates final FW temperature control during power operation and power maneuvering. It comes in-service gradually from 70-90% power. The final stage HP FW heater is fully in-service to raise FW temperature when power is above 90% to increase stability margins.

The main FW line also connects to the long cycle cleanup recirculation path line back to the main condenser.

A final FW strainer is installed to minimize debris entry into the NBS. The strainer is placed downstream of FW Heater No. 6 and upstream of the FW measurement devices. Downstream of the HP FW heaters, the FW line divides into two (2) FW lines that pass through the containment via a system and CIV. Inside containment, each of these lines splits again into a total of four (4) lines that connect to the reactor via the NBS FW reactor isolation valves.

The SDC (Chapter 9A, Subsection 9A.2.3) return lines connect to the FW lines upstream of the outboard CIVs. SDC uses the CFS containment penetrations to reduce the total number of containment penetrations.

The CFS design follows recommended practices for prevention of water damage to steam turbines and prevention of excessive corrosion/erosion damage due to flashing liquid flow (Subsection 10.2.3.3). The system capacity allows for continuous long-term power plant operation with either an HP FW heater or LP FW heater out of service.

#### **10.3.2.3.1 Component Description**

The following paragraphs provides descriptive information specific to the major equipment in the CFS. Table 10.3.2-1 summarizes major components in the CFS.

#### **Condensate Pumps**

The CFS consists of two (2) 100% capacity condensate pumps.

The condensate pumps are located below the normal water level of the main condenser hotwell to ensure the pump suction remains flooded. Each condensate pump has an individual suction line and is equipped with an anti-vortex device. A permanent suction strainer is provided on the suction of the condensate pumps to remove larger debris from the condensate to protect the pumps.

A condensate pump discharge header connects the CFS to the CFD (Subsection 10.3.1).

#### **Low Pressure Feedwater Heaters**

The stage 1 and 2 duplex LP FW heaters are located within each condenser neck.

LP FW Heaters No. 1, No. 2, and No. 3 are closed FW heater types, designed in accordance with ASME BPVC, Section VIII, Division 1 (Reference 10.3.2-2), and the Heat Exchange Institute Standards.

The FW heater tubes and tubesheets are corrosion resistant stainless steel. Tube to tubesheet joints are welded. The FW heater internal vent piping is also constructed of corrosion resistant stainless steel. The FW heater shell material considers the effects of corrosion product metals to minimize deposits on fuel.

FW Heater No. 3 is designed for 100% of the plant condensate flow and duty. FW Heater No. 1 and No. 2 are designed for 50% of the plant condensate flow and duty.

### **High Pressure Feedwater Heaters**

Three (3) HP FW heaters are provided. The HP FW heaters are horizontal U-tube type heat exchangers with integral drain cooler. Each HP FW heater is designed for 100% of the plant FW flow.

The FW heater tubes and tubesheets are corrosion resistant stainless steel. Tube to tubesheet joints are welded. The FW heater internal vent piping is also constructed of corrosion resistant stainless steel. The FW heater shell material considers the effects of corrosion product metals to minimize deposits on fuel.

The FW heaters are designed in accordance with ASME BPVC, Section VIII, Division 1 (Reference 10.3.2-2), and Heat Exchange Institute Standards.

Downstream of the HP FW heaters, the FW line divides into two (2) FW lines that pass through the containment via a system and CIV. Inside containment, each of these lines splits again into a total of four (4) lines that connect to the reactor via the NBS FW reactor isolation valves. The main FW line also connects to the long cycle cleanup recirculation path line back to the main condenser.

### **Feedwater Pumps**

Two (2) FW pumps, driven by an electrical motor and powered by an Adjustable Speed Drive, are sized for 100% of the total rated FW flow, plus margin for control. One (1) pump is in operation while the other is on standby.

The FW pumps are located on grade elevation of the TB.

A bypass is provided around the FW pumps to permit FW supply to the reactor during early startup using only the condensate pumps as well as during preparation for outage leakage tests. During startup, a Low Flow Control Valve is used to control the RPV level.

### **Piping and Valves**

The CFS CIVs and piping are SC1 and are designed to ASME BPVC, Section III, Class 2 (Reference 10.3.2-3).

To ensure containment integrity at the primary vessel containment penetrations, the containment piping and isolation valves, and the CFS piping upstream of the CIVs to and including the system isolation valves are designed to Seismic Category A; Boiler and Pressure Vessel Code, Section III, Division 1-Subsection NCD; and Quality Group B requirements.

The CFS piping upstream of the system isolation valves is designed SC1, ASME Section III, Class 2 up to and including the seismic restraint upstream from the system isolation valve. The CFS piping upstream from the seismic restraint, located upstream of the system isolation valve, is classified as SC3, Quality Group D, Non-Nuclear Seismic Category, and designed to meet the requirements of ASME Code for Pressure Piping B31.1, Power Piping (Reference 10.3.2-4).

The CFS piping and valves are constructed of corrosion resistant stainless steel.

The low flow level control valve is capable of controlling condensate flow smoothly at low power levels.

The layout of the CFS and the characteristics of the valves is such that water hammer loads are below the reactor design limits. The design of the CFS complies with NUREG-0927 (Reference 10.3.2-5).

The normal and alternate drain control valves on the shell side of the FW heaters are located to minimize flashing upstream of the control valve under any load or drain flow condition and materials are selected to be compatible with any flashing conditions.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSC.

#### **10.3.2.4 Materials**

The CFS is designed to avoid FAC damage. The design and layout of the piping system considers the effects on the piping material of fluid velocity, bend location and the location of flash points. The selection of piping layout, velocity and material is based on successful operating experience.

#### **10.3.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.3.2-2 for system interfaces.

#### **10.3.2.6 System and Equipment Operation**

##### ***10.3.2.6.1 Normal Operational Concept***

##### **Startup Mode**

The CFS is started by first placing the condensate portion of the system in service. Condensate is sent through the CFD and then returned to the main condenser through the condensate pump minimum flow recirculation line for short cycle cleanup.

The CFD is aligned in the bypass mode and the condensate pump minimum flow control valve is opened. The condensate pumps are started and the CFD is placed in-service before the CFD bypass valve is closed. The minimum flow bypass valve modulates as the proper flow rates and pressure drops are established in the CFD. Condensate from the condenser hotwell is recirculated through the auxiliary condensate loop and min flow/cleanup line until the required water quality is achieved. Condensate may be recirculated through the short cycle cleanup line to establish condensate quality in the hotwell.

When the required condensate quality has been achieved, flow is directed through the LP and HP FW heaters and associated piping by establishing flow through the long cycle cleanup line.

A bypass is provided around the FW pumps to permit FW supply to the reactor during early startup using only the condensate pumps as well as during preparation for outage leakage tests. During startup, a Low Flow Control Valve is used to control the RPV level.

FW pumps may be started when the required condensate purity levels are reached, and adequate pressure is provided to the FW pump suction. FW flows through the HP FW heaters and back to the condenser through the FW pump recirculation valve in long path cleanup mode.

The FW cleanup flowpath is used to recycle the condensate and FW through the CFD before admission to the reactor core.

Normal and alternate drain line control valves modulate as required to maintain water levels in the closed FW heaters based on the signals from the corresponding level transmitters.

##### **Recirculation/Cleanup Mode**

The CFS is designed to enable the entire condensate and FW water volume (including the hotwell and LP & HP FW heaters) to be recirculated via the CFD before plant startup, ensuring proper water quality before admission to the reactor.

The CFS cleanup process is carried out using the condensate pumps to clean the condensate subsystem and using the FW pumps operating at lower speed to clean the FW subsystem.

The water inventory in the CST may also be cleaned by circulating through the hotwell using the hotwell makeup and condensate reject lines.

### **Normal Plant Operation**

The CFS is normally operated with one (1) condensate pump operating and one (1) on standby and one (1) FW pump operating and one (1) on standby. All LP and HP FW heater isolation valves are normally open. The LP FW heaters bypass valves and the HP FW heaters bypass valves are normally closed.

During power operation, reactor vessel water level is controlled by the Reactor Level Control System (RLC) (Chapter 7, Subsection 7.3.3.2). At higher powers, typically above 25%, RLC controls FW flow and thus reactor water level by adjusting FW pump speed. At lower powers the RLC controls the position of the Low Flow Control Valve based on target reactor water level.

The CFS continuously drains the condensed HP and LP extraction steam in a cascading setup from FW Heater No. 6 to FW heaters 1 A/B and then the condenser hotwell. Normal and alternate drain line control valves will maintain water level in each HP and LP FW heater based on the signal from the corresponding level transmitters.

The CFS continuously vents vapor and non-condensables from the closed FW heaters to the main condenser.

#### **10.3.2.6.2 Off-Normal Operational Concept**

### **Condensate Pump Trip**

The CFS is normally operated with one condensate pump operating and one on standby. In the event of a condensate pump trip, the standby pump starts to continue supplying condensate to the FW pump suction and RPV. The CFS has built-in logic to immediately start the standby pump to minimize the system transients to the system and RPV.

### **Feedwater Pump Trip**

The CFS is normally operated with one FW pump operating and one on standby. In the event of a FW pump trip, the standby pump starts to continue supplying FW to the RPV. The CFS has built-in logic to immediately start the standby pump to minimize the transients to the system and RPV.

### **One LP or HP Feedwater Heater Out of Service or Isolated**

The CFS can be operated with one LP FW heater or one HP FW heater out of service by closing the FW heater inlet and outlet isolation air-operated valves and opening the air-operated FW heater bypass valve. The FW heater bypass valve is fully opened prior to isolation of one of the LP FW heaters or HP FW heaters.

The Extraction Steam System (Subsection 10.2.3.11) HP or LP extraction isolation valves for the affected heater close, the non-return valves for the affected heater (release to close), and the extraction steam drain isolation valves for the affected heater open, providing a path to the main condenser. The Extraction Steam System supplies steam to Heaters No. 1 A/B through Heater No. 5.

The steam supply for Heater No. 6 is from the MS subsystem (Section 10.4). When the heater is out of the service, the steam supply isolation valve closes and the steam drain isolation valves open, providing a path to the main condenser.



The CFS provides signals indicating that valves are closed to the SC2 and SC3 I&C System to provide automatic reactor power reduction upon isolation of one of the LP FW heaters or HP FW heaters.

### **10.3.2.7 Instrumentation and Control**

This section describes some I&Cs required for system operation and control.

Instrumentation is provided to verify condensate and FW pumps, as well as FW heater performance. Provisions include instrumentation to provide input to the plant heat balance.

Differential pressure transmitters are provided to monitor the following parameters:

1. LP and HP FW heater differential pressure
2. Differential pressure across the FW Low Flow Control Valve

Flow transmitters are provided to monitor the following parameters:

1. The discharge flow in the condensate pump header. The measurement is used to modulate the valve on the minimum flow/short cycle cleanup line. It is also used to trip the pump when flow through the pump is below a preset value.
2. Flow is measured at each main FW pump discharge and each main FW pump minimum flow recirculation line. This allows for measurement of the total FW flow from each pump.
3. The FW supply line to the reactor water flow. The measurement is used to determine the FW supply to the reactor. The signals from the flow transmitter are used by the FW Control System to control reactor level by adjusting the FW pump speed.

Pressure transmitters are provided to monitor the following parameters:

1. Suction and discharge pressure of each condensate pump
2. Pressure at the inlet of the auxiliary condensate loop.
3. The suction pressure and discharge pressure of each FW pump. The suction flow pressure measurement is also used to control the corresponding pump minimum flow control valve and to alarm and trip the pump when the pressure is below a preset minimum value
4. Pressure for the FW supply line to the reactor (each branch)
5. Normal drain outlets of the closed FW heaters
6. FW heater shell pressure.

Temperature is monitored in the following system locations:

1. Each Condensate pump suction and discharge
2. Each FW heater inlet and outlet line
3. FW temperature in each FW line to the reactor. These temperature measurements are provided in each FW line to the reactor to determine the reactor FW inlet temperature and to monitor the temperature difference. These temperature measurements are also used as part of the control scheme to control the final FW temperature

Vibration transducers and monitoring equipment are provided for the condensate pumps and FW pumps in accordance with API 670 (Reference 10.3.2-6). Vibration level control limits are in accordance with API 610 (Reference 10.3.2-7).

#### **10.3.2.8 Monitoring, Inspection, Testing, and Maintenance**

The equipment and components of the CFS are designed to support inspection and maintenance activities.

Routine testing of the CFS is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. Suitable access and/or remote functions to permit in-service testing and inspection of the FW piping are provided. ISI accessibility is provided by appropriate arrangement of piping and major equipment.

For the portion of the system that is ASME Class 2, accessible arrangement of vents and drains in the system comply with ASME Code Section XI (Reference 10.3.2-8) requirements for the performance of ISI and testing for assessing operational readiness.

CFS piping and valves for the containment penetrations are tested in accordance with 10 CFR 50, Appendix J and REGDOC-2.5.2 (Reference 10.3.2-1). Test and vent connections are provided at the CIVs to validate local leak rate limits.

#### **10.3.2.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.3.2.10 Performance and Safety Evaluation**

The CFS is capable of performing its design functions as presented in Subsection 10.3.2.1, during the modes of operation when the system is intended to function. To ensure containment integrity at the Steel-plate Composite Containment Vessel penetrations, the CIVs, and piping are designed to Seismic Categories A and B respectively; and to the requirements of the ASME BPVC, Section III, Division 1-Subsection NCD.

#### **10.3.2.11 References**

- 10.3.2-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.3.2-2 ASME BPVC-VIII, "BPVC Section VIII-Rules for Construction of Pressure Vessels Division 1," American Society of Mechanical Engineers.
- 10.3.2-3 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 10.3.2-4 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 10.3.2-5 USNRC NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants."
- 10.3.2-6 API 670, "Machinery Protection Systems," American Petroleum Institute.
- 10.3.2-7 API 610, "Centrifugal Pumps for Petroleum, Petrochemical and Natural Gas Industries," American Petroleum Institute.
- 10.3.2-8 ASME BPVC-XI-1, "BPVC Section XI-Rules for Inservice Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers.

**Table 10.3.2-1: Condensate Feedwater Heating System Major Components**

<b>Condensate Pumps</b>	
Number of Pumps	2 (1 normally operating)
Pump Type	Fixed speed, centrifugal
Driver Type	Motor
<b>Low Pressure Feedwater Heaters</b>	
No. 1 and No. 2:	
Number per stage	1 duplex per stage for a total of 2
No. 3	
Number per stage	1
<b>High Pressure Feedwater Heaters</b>	
No. 4	
Number per stage	1
No.5:	
Number per stage	1
No.6:	
Number per stage	1
<b>Reactor Feedwater Pumps</b>	
Number of Pumps	2 (1 normally operating)
Pump Type	Variable speed, centrifugal
Driver Type	Motor

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.3.2-2: Condensate Feedwater Heating System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	CFS provides FW to NBS at required flow rate, pressure, and temperature. CFS receives MS from the NBS to the HP FW Heater No. 6	At the inlet of the outboard reactor vessel isolation valve At the inlet of the temperature control valve
SC1 Instrumentation and Control System	Provides CFS Containment Isolation signals and FW isolation signals, and receives signals from SC1 instrumentation	SC1 I&C System DCIS termination cabinet
SC2 and SC3 Instrumentation and Control System	Provides inputs to and receives signals from the SC2 and SC3 I&C System	SC2 and SC3 I&C System DCIS termination cabinet
Non-Safety Instrumentation and Controls	Non-Safety I&Cs provides Non-Safety I&C Control for pumps, valves, and instrumentation	Non-Safety I&C System DCIS termination cabinet
Process Radiation and Environmental Monitoring System	Samples taken continuously to monitor water quality	Second isolation valve from CFS process stream
CRD System/HP Injection System and Hydraulic Control Units	CFS provides primary source of treated water to the CRD pump suction	Outlet of air-controlled isolation valve
Shutdown Cooling	The shutdown cooling return lines tie into the FW lines upstream of the outboard CIV	At the inlet of the AOVs
Reactor Water Cleanup System (CUW)	CFS provides cooling water for CUW regenerative heat exchanger and receives water from CUW for processing through CFD	At the inlet and outlet of the CUW heat exchanger At the outlet of the CUW isolation valve
Liquid Waste Management System	CFS provides condensate reject to the CST	At the inlet to the CST isolation valve

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Offgas System	CFS provides cooling to offgas condenser	At the inlet and outlet of the offgas condenser
Condensate Filters and Demineralizers	CFS provides condensate water for purification and receives condensate after purification	At the inlet and outlet of the CFD header
Main Turbine Equipment	MTE provides steam to the FW heaters. CFS provides cooling to gland seal condenser	At the inlet to the FW heaters from the MTE extraction system At the inlet and outlet of the gland seal condenser
Moisture Separator Reheater System	MSR drains to shell side of the FW heaters	At the inlet to the FW heaters from the MSR drains
Main Condenser and Auxiliaries	MCA provides condensate source. CFS provides hood spray and cooling to SJAE intercondensers	All connections to MCA are located at the outlet/inlet with the main condenser At the inlet and outlet of the SJAE Intercondensers A/B
Plant Cooling Water	Plant Cooling Water (PCW) provides cooling water to the condensate pump and FW pump motors	At the pump cooling water inlet and outlet
Plant Pneumatics System	Provides control air for valves	At the inlet to the CFS AOVs
Hydrogen Water Chemistry	CFS provides injection port(s) for Hydrogen Water Chemistry	The Hydrogen Water Chemistry interfaces at the outlet of the isolation valves in Hydrogen Water Chemistry
Online NobleChem™	CFS provides injection port(s) for Online NobleChem™	Online NobleChem™ interfaces downstream of the isolation

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
		and root valves to a shared flange with CFS
SC2 and SC3 Electrical Distribution System	SC2 and SC3 Electrical Distribution System provides SC low and medium voltage power	Electrical terminals of the power cables at the load end
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category low and medium voltage power	Electrical terminals of the power cables
Equipment and Floor Drain System	EFS receives discharged water from CFS drains	At the outlet of drain valves to EFS

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

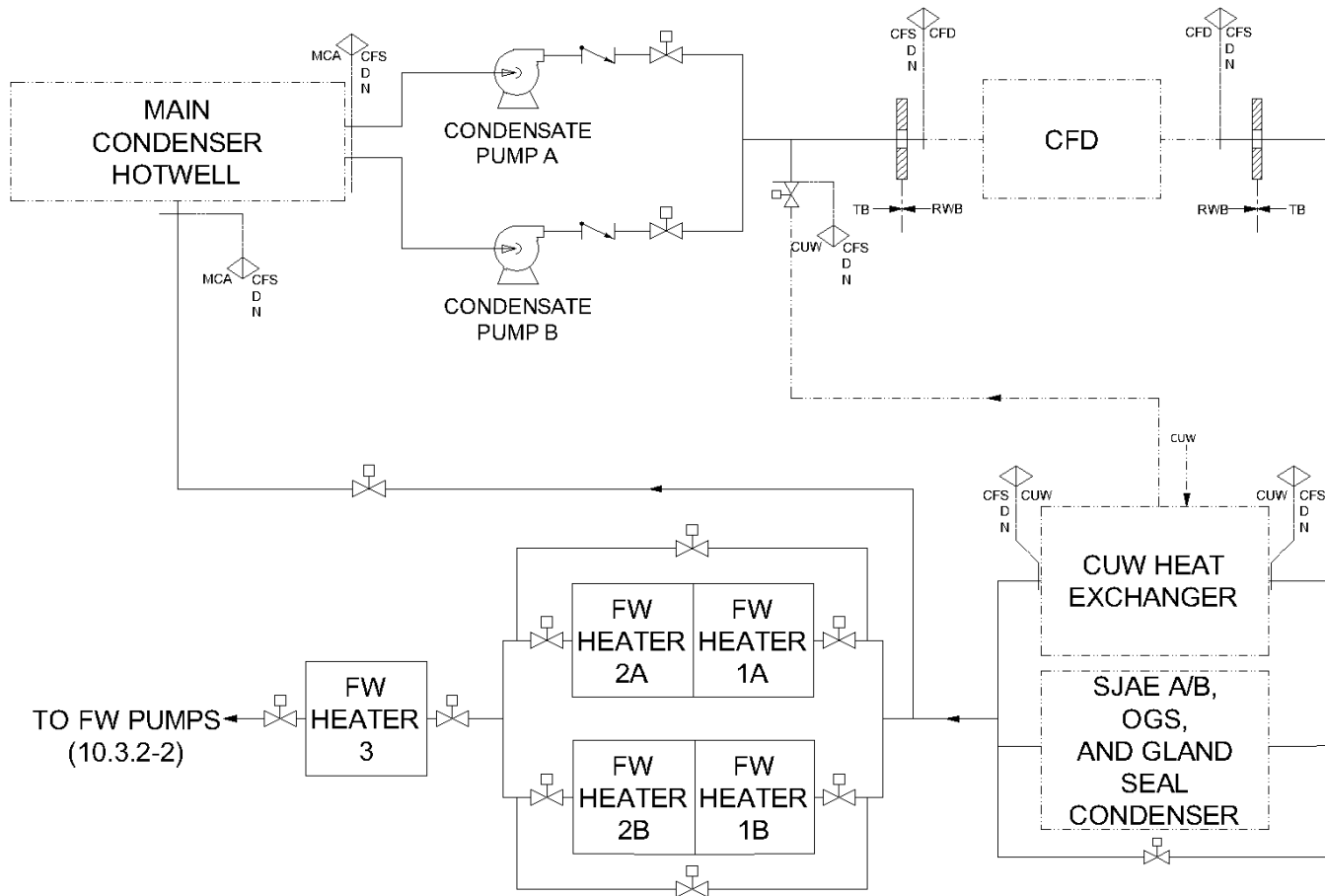


Figure 10.3.2-1: Condensate Portion of Condensate and Feedwater Heating System

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

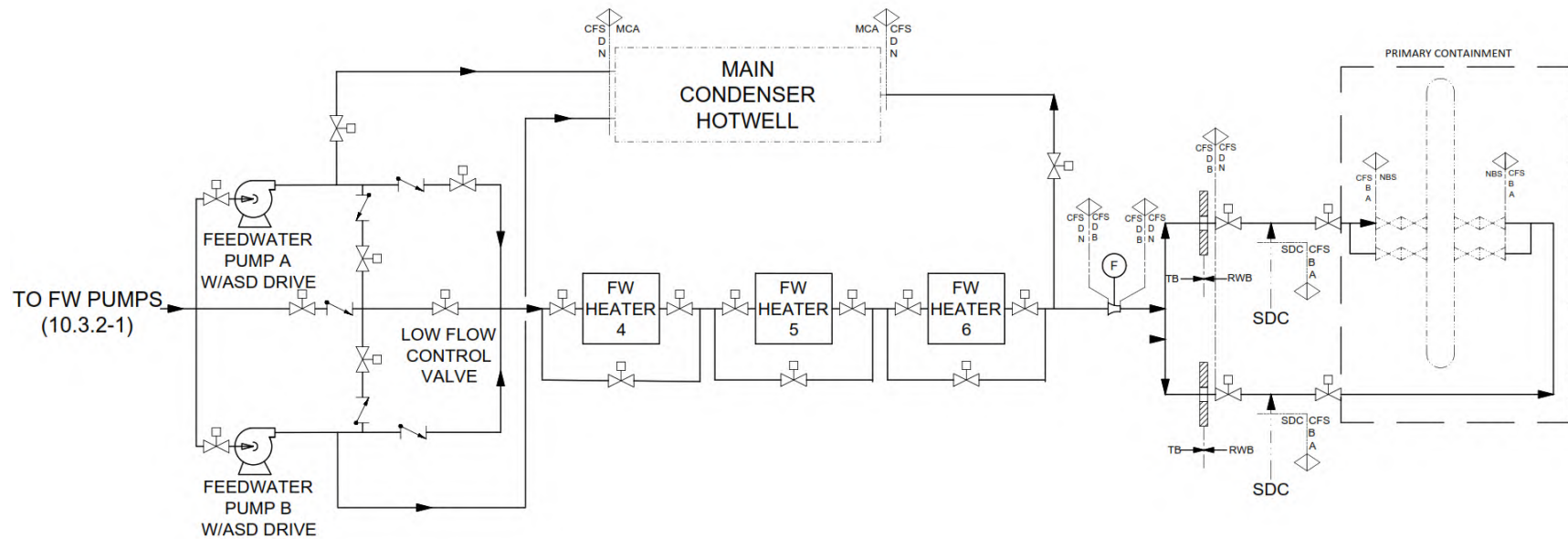


Figure 10.3.2-2: Feedwater Portion of Condensate and Feedwater Heating System



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

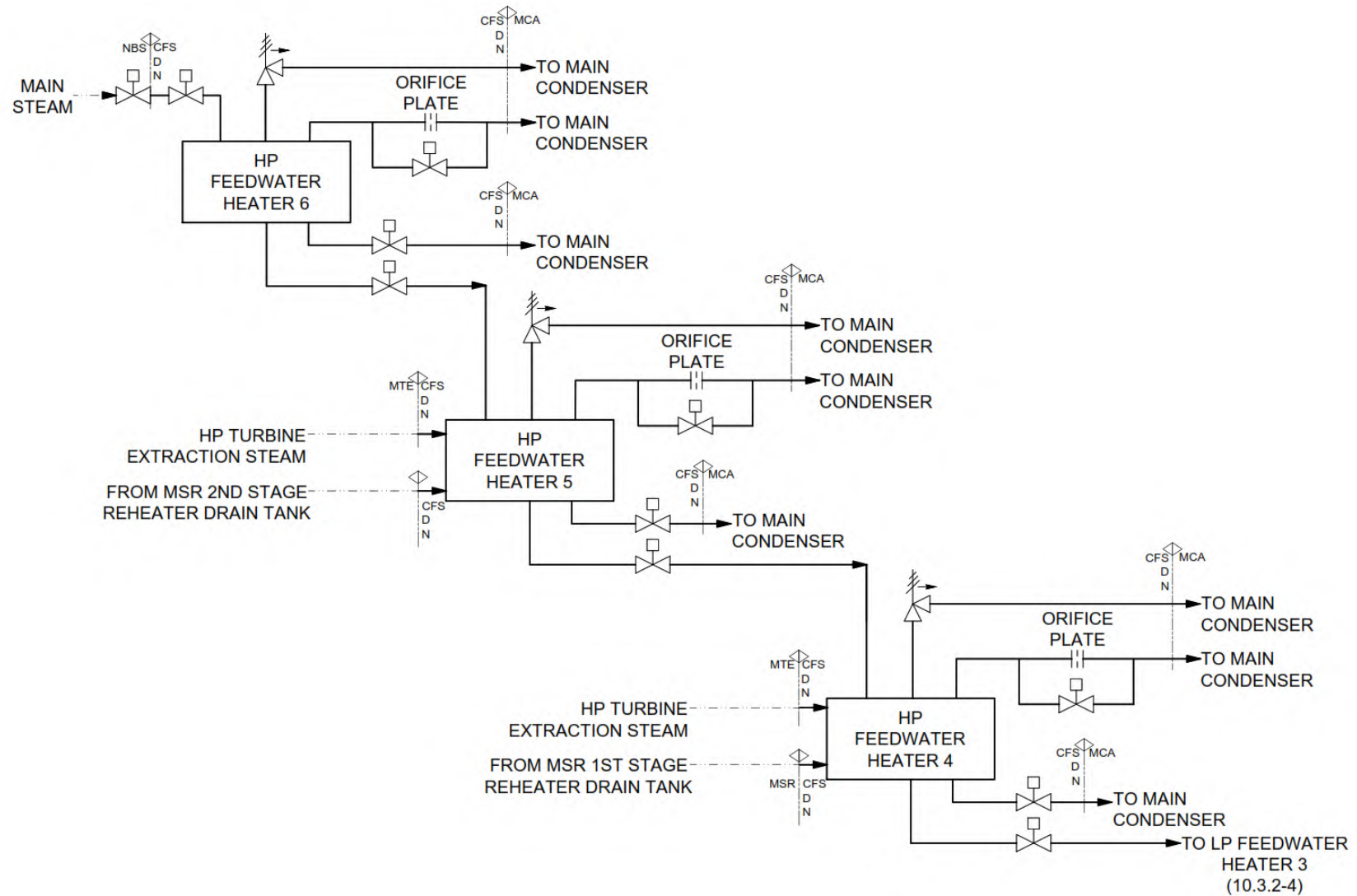


Figure 10.3-2-3: Vents and Drains for High Pressure Heaters in Condensate and Feedwater Heating System

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

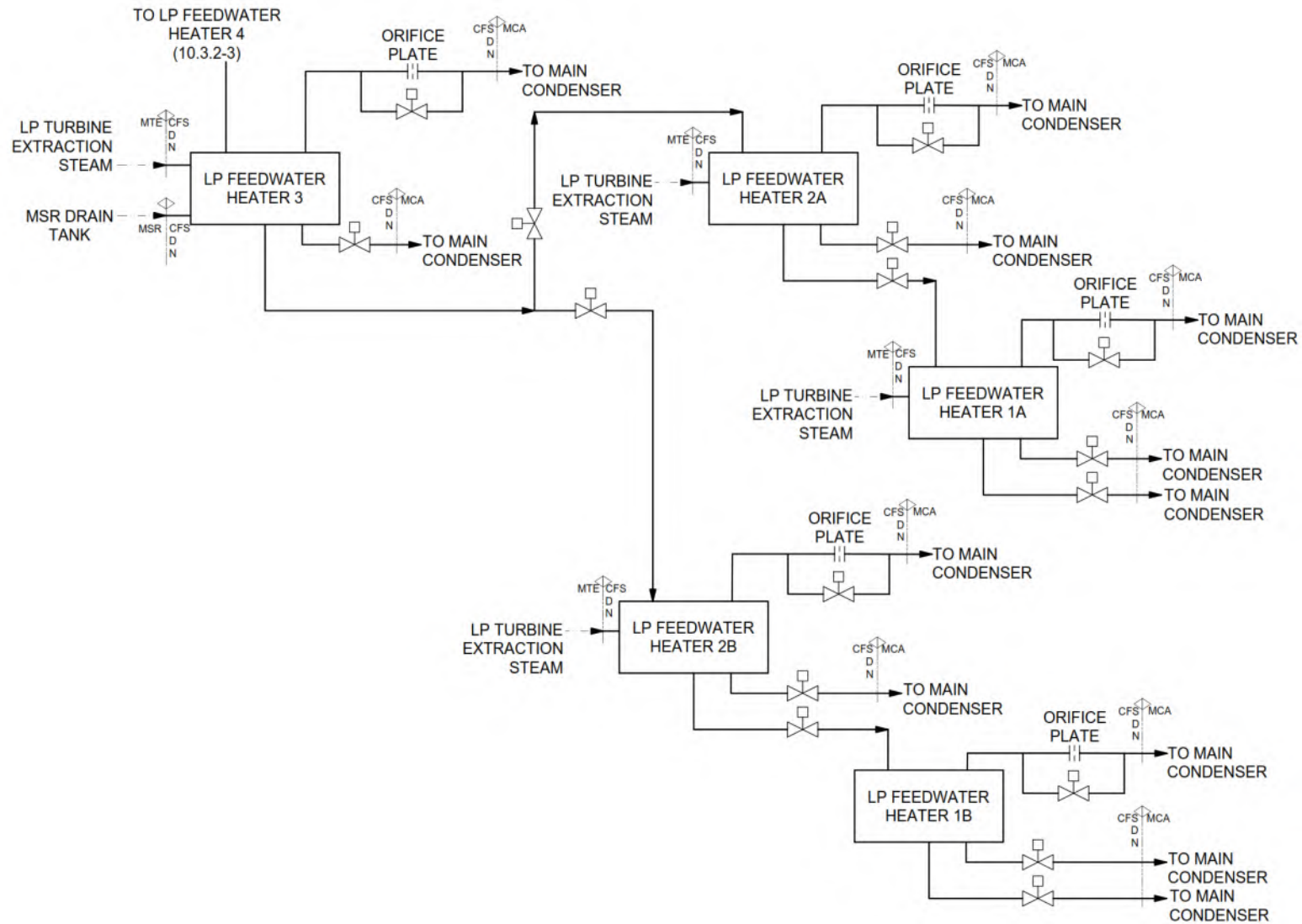


Figure 10.3.2-4: Vents and Drains for Low Pressure Heaters in Condensate and Feedwater Heating System

## **10.4 Main Steam System**

The MS is a subsystem of the NBS. This section describes the MS subsystem from the non-safety ASME B31.1 piping interface with the ASME BPVC, Section III, Class 2 piping downstream of the SC1 Seismic Interface Restraint (SIR), to the TSVs (Subsection 10.2.3.1), the TBVs, the MSL drains, and other load isolation/maintenance valves. The lines to these loads, all connecting branch lines up to and including their respective isolation valves, and all associated piping supports are also part of the MS subsystem.

Information pertaining to the MS subsystem from the outboard side of the outboard MS Reactor Isolation Valves up to and including the SIR is addressed in Chapter 5.

### **10.4.1 System and Equipment Functions**

The system and equipment functions associated with the MS subsystem are identified below.

#### **10.4.1.1 Normal Functions (Non-Safety Category)**

The MS subsystem described in this section performs the following Non-Safety Category functions:

1. As a subsystem of the NBS, it contributes to the ability to produce the steam flow capacity and conditions to the main turbine for electric power production.
2. As a subsystem of the NBS, it contributes to the transport of steam generated in the RPV to the main turbine for electric power production in Mode 1.

#### **10.4.1.2 Normal Functions (Safety Category)**

The MS subsystem described in this section does not perform any Safety Category functions during normal conditions.

#### **10.4.1.3 Off-Normal Functions (Non-Safety Category)**

The MS subsystem described in this section does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.4.1.4 Off-Normal Functions (Safety Category)**

The MS subsystem described in this section does not perform any Safety Category functions during off-normal conditions.

### **10.4.2 Safety Design Bases**

The design of the MS System meets CNSC requirements specified in REGDOC-2.5.2 Section 8.3 as it relates to the steam lines, Section 6.1 (Reference 10.4-1) as it relates to design for Defence-in-Depth and Chapter 7, Section 7.1 as it relates to classification of SSC.

### **10.4.3 Description**

Figure 10.4-1 depicts the MS subsystem.

This section describes the portion of the MS subsystem located downstream of the ASME BPVC, Section III, Class 2 piping interface with the ASME B31.1 piping. Two MSLs are routed to the TSVs, the equalizing header, and the MSL drains.

The MS equalizing header is designed to be located as close as practical to the TSVs and at least as large in diameter as a single main steam line. The MS equalizing header supplies high pressure steam via branch piping to the TBVs, the MSR, the FW Heater No. 6, and the TASS.

The MSL drains drain condensate from the MSLs to the main condenser during startup, low power operation, normal power operation, and shutdown. A reduction in power below 40% Rated

Thermal Power (RTP) results in the automatic opening of drain line valves, thereby establishing drain flow to the main condenser.

Subsections 10.7.6 and 10.2.6 provide information related to RPC during startup and normal operation respectively. In addition, Chapter 6, Subsection 6.2.2 provides information pertaining to overpressure protection.

#### **10.4.3.1 Component Description**

##### **10.4.3.1.1 Main Steam Lines**

MSLs are designed to mitigate acoustic loads generated as a result of NBS and interfacing system configuration such that their impact on system functional capabilities and material condition is acceptable.

The MS piping in the TB is designed to ASME B31.1 (Reference 10.4-2), Non-Nuclear Seismic Category requirements including all the connecting branch lines up to the various steam loads including their respective isolation valves.

The MS subsystem piping has a design pressure of 10.342 MPaA (1500 psig) and design temperature of 314.4°C (598°F). It is sized to limit steam velocities at full power steady-state operating conditions (to limit the effects of flashing, noise, vibration, water/steam hammer, and erosion).

The MS system is designed so that the pressure drop through the piping at 100% RTP condition does not exceed the minimum and maximum limits.

NBS is designed so that the pressure downstream of the outboard MSCIV is equal to or greater than the pressure specified for standard plant operating conditions with 10% margin.

##### **10.4.3.1.2 Main Steam Line Drains**

The MS drain lines provide the ability to drain the condensate from the MSL to the condenser in a controlled manner during startup, lower power operation (approximately 40% RTP for the bypass drain valves), normal power operation, and shutdown.

Drains are at low points or catch-points to drain condensate back to the main condenser.

The drain line isolation valves on MSL drain lines fail open. The drain lines contain an orifice and an in-line strainer.

#### **10.4.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from corrosion (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

#### **10.4.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.4-1 for system interfaces.

#### **10.4.6 System and Equipment Operation**

Chapter 5, Subsection 5.3.3 describes NBS operations. The subsections below focus on operations of the MS subsystem in each of the facility modes.

##### **10.4.6.1 Initial Configuration (Pre-Startup)**

System configuration is established per plant procedures.

#### **10.4.6.2 System Startup**

Chapter 5, Subsection 5.3.4.2 describes NBS startup. The MSL drains remain open to approximately 40% RTP to remove excess moisture carryover during NBS startup operations.

#### **10.4.6.3 Normal Operations**

Chapter 5, Subsection 5.3.4.3 describes normal operations of the NBS system. During normal operations, the MS subsystem continues to transport steam generated in the RPV that passed through containment from the RPV nozzles through the MS Reactor Isolation Valves and MSCIVs to provide the steam power needed to drive the main turbine. Steam is directed through the open maintenance block valves to the MSR, FW Heater No. 6, and TASS. MSL drain line valves open and close as a function of the level of condensed steam in the drain pot up to about 40% power where they are closed.

During power operation, the NBS also vents non-condensable gases generated in the RPV through the RPV head vent to the MS piping.

#### **10.4.6.4 Off-Normal Operations**

Chapter 5, Subsection 5.3.4.4 describes planned and unplanned transients for the NBS system. Transients that lead to reductions in power to 40% or less result in automatic opening of the drain line valves to the main condenser.

Unplanned transients are listed in the fault list described in Chapter 3, Appendix 3A.

#### **10.4.6.5 System Shutdown**

Chapter 5, Subsection 5.3.4.5 describes NBS system shutdown consisting of hot shutdown, stable shutdown, and cold shutdown. Normally, the reactor design is to be cooled down from the hot shutdown condition by opening one or more TBVs to direct steam to the main condenser. Opening of a TBV reduces pressure in the reactor and the steam lines. The reactor coolant is flashed to steam, which flows through the MSLs and the TBVs and ultimately to the main condenser.

The MS drain lines provide the ability to drain the condensate from the MS lines to the condenser in a controlled manner during startup, lower power operation (approximately 40% nuclear boiler rated for the drain valves), normal power operation, and shutdown.

#### **10.4.7 Instrumentation and Controls**

The I&Cs for the MS subsystem within the NBS described in Chapter 5 consists of the following:

1. Chapter 7, Subsection 7.3.3 describes the I&Cs for the RPC system.
2. The TB contains MSL leak detection instrumentation. This instrumentation is designed to fail-safe as the TB is a Non-Nuclear Seismic structure.
3. The MSL drain isolation valves have control logic interlocked to support automatic plant startup and shutdown operation.
4. Pressure transmitters are provided and located on the MSLs near the equalizing header in the TB to monitor steam line parameters.

#### **10.4.8 Monitoring, Inspection, Testing, and Maintenance**

Piping and components are inspected and tested in accordance with the requirements of ASME B31.1 (Reference 10.4-2). Areas requiring inspections are provided with access spaces/routes and with removable and reusable insulation coverings.

The MS equalizing header arrangement upstream of the TSVs allows for online testing.

#### **10.4.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.4.10 Performance and Safety Evaluation**

Components and piping for the MS subsystem described in this section are designed in accordance with the ASME B31.1 code (Reference 10.4-2). MSLs are designed to ensure there are no negative interactions between lines. This ensures that the MS subsystem can accommodate operational stresses resulting from static and dynamic loads, including water (steam) hammer and includes provisions to limit water entrainment. Operating and maintenance procedures include adequate precautions to minimize the potential for water (steam) hammer.

The break of a MSL does not result in offsite radiation exposures in excess of the site limits as discussed in Chapter 15, Subsection 15.5.9.2.1.

#### **10.4.11 References**

- 10.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.4-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.4-1: Main Steam System Subsystem Interfaces**

Interface System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Control System	The Non-Safety DCIS along with plant automation functions provide a distributed control and instrumentation data communication network to support non-safety portion of MS subsystem.	Interface is at the Non-Safety DCIS termination cabinet
Process Radiation and Environmental Monitoring System	In the event of a large steam line pipe break outside containment, the plant's PREMS is designed to continuously monitor to detect such an occurrence.	PREMS Termination Cabinets
Condensate and Feedwater Heating System	MS subsystem provides 32.54 kg/s (0.2583 Mlbm/hr) steam from the MSLs to the FW Heater No. 6.	NBS block valve to CFS piping to FW Heater No. 6
Main Turbine Equipment	The NBS supplies steam to the main turbine during power operation and transients such as load variations.	MSL to TSVs connection
Main Turbine Equipment	The Equalizing Header is sized so as to permit the performance of a full close test at 90% power of one of the main TSVs during operation without imposing a severe pressure load transient on the reactor vessel internals.	MSL to TSVs connection
Moisture Separator Reheater System	MS supplies 21.62 kg/s (0.1716 Mlbm/hr) of steam for the MSR System.	NBS block valve to MSR piping to MSR
Turbine Bypass System	MS supplies steam to the TBVs via piping between the equalizing header and the TBVs.	MSL piping to TBV A and TBV B
Main Condenser and Auxiliaries	During startup and up to approximately 40% power operation, the main condenser via the MS drain lines receives steam, steam condensate, and non-condensable gases from the RPV and MSLs.	NBS block valve to MCA piping to condenser & NBS block valve to MCA piping to condenser

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Interface System	Interface Description	Interface Boundary
Non-Safety Electrical Distribution System	The Non-Safety Electrical Distribution System provides power to the SCN valves in the MS subsystem. These valves are all located in the TB.	The interface point will be the electrical terminations of the power cables at the valve actuator
Equipment and Floor Drain System	The EFS provides sumps or drain tanks to the NBS system.	Interface is the connection from the NBS to the EFS in the TB
Turbine Building Structure	The TB provides the necessary space and structural support for the NBS MSLs and MS drain lines and components.	Interfaces are the pipe support attachments to the TB structures



The diagram illustrates the main steam system (MSS) configuration. It features two parallel steam generators, each with its own set of control valves and instrumentation. The system is connected to a common main steam header, which then branches out to various components including the RB and TB systems, the STEAM TUNNEL, and the main steam header itself. Key components include:
 

- Steam Generators:** Two parallel units, each with a main steam drain (MS DRN) and a main steam header connection.
- Valves:** MS DRN, TSV A, TSV B, and various control valves (e.g., TBP, TBV A, TBV B).
- Pumps and Flow Control:** NBS (Normal Boiling System), MCA (Main Control Actuator), and EFS (Emergency Flow Stop).
- Sensors:** Temperature (T) and Pressure (P) sensors.
- Connections:** Connections to the RB and TB systems, the STEAM TUNNEL, and the main steam header.

10-70

## **10.5 Main Condenser and Auxiliaries System**

The MCA System is the heat sink for the power generation and normal reactor cooldown and plant startup activities. The MCA System consists of the main condenser, two SJAE skids, and two condenser vacuum pump skids as well as associated piping, valves, instrumentation, and controls.

### **10.5.1 System and Equipment Functions**

The MCA performs the function listed below during normal and off-normal conditions.

#### **10.5.1.1 Normal Functions (Non-Safety Category)**

The MCA provides the following Non-Safety Category functions during normal conditions:

1. The MCA System provides a main condenser with interfaces to other systems as needed to condense steam exhaust from the main turbine and main turbine bypass.
2. The main condenser includes a reservoir (hotwell) in which the condensate is collected as a suction source for the condensate pumps.
3. The main condenser collects equipment drains, vents, and relief valve discharges from other plant equipment.
4. The condenser vacuum pumps draw a vacuum and remove non-condensable gases from the condenser shell during startup.

#### **10.5.1.2 Normal Functions (Safety Category)**

The MCA provides the following Safety Category functions during normal conditions:

1. The MCA provides the capability to measure main condenser vacuum (pressure).
2. The MCA provides the heat sink to condense reactor steam or drainage from the FW heaters and other steam supply users.
3. The SJAE draws a vacuum and removes non-condensable gases from the condenser shell.
4. The MCA provides a condensate supply to the condensate pump suction.

#### **10.5.1.3 Off-Normal Functions (Non-Safety Category)**

The system does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.5.1.4 Off-Normal Functions (Safety Category)**

The system does not perform any Safety Category functions during off-normal conditions.

### **10.5.2 Safety Design Bases**

The safety design bases for the MCA include the items listed below. Codes and standards related to pressure boundary are identified in the component descriptions in Subsection 10.5.3.1 as applicable. MCA System equipment is categorized as Non-Nuclear Seismic.

**SC2 Requirements:**

1. As part of Defense Line (DL) 4a, the MCA has the capability to measure main condenser vacuum (pressure) to initiate a scram when the mode switch is in the RUN position, due to low condenser vacuum.

**SC3 Requirements:**

1. As part of DL2, the MCA provides the capability to measure main condenser pressure to initiate TBV closure, due to low condenser vacuum.
2. The MCA provides the heat sink to condense reactor steam or drainage from the FW heaters and other steam supply users.
3. The SJAЕ draws a vacuum and removes non-condensable gases from the condenser shell.
4. The MCA provides a condensate supply to the condensate pump suction.

The MCA is not required to operate during or after a design basis event. The design of the MCA meets CNSC requirements specified in REGDOC-2.5.2 Section 6.1 (Reference 10.5-1) as it relates to design for Defence-in-Depth and Section 7.1 as it relates to classification of SSC.

### **10.5.3 Description**

The main condenser is a single-pressure, two-shell, unit. Each shell is located beneath its respective LP turbine. The two condenser shells operate at similar pressures and drain to hotwells that are cross connected. Circulating water flows through each of the two single-pass tube bundles to condense the turbine exhaust steam into the hotwells. The main condenser receives and condenses turbine exhaust steam and turbine bypass steam during all modes of operation. The main condenser provides hold-up for N16 decay, and supplies condensate to the condensate pumps. The main condenser also serves as a collection point for other steam cycle miscellaneous drains, vents, and relief valve discharges.

Two 100% capacity SJAEs are used to maintain the turbine backpressure and remove non-condensable gases from the main condenser. Non-condensable gases extracted from the condenser are exhausted to the Offgas System.

During startup, two condenser vacuum pumps draw the initial condenser vacuum and exhaust the gases to the TB Heating, Ventilation, and Air Conditioning (Chapter 9A, Subsection 9A.5.4).

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSC.

#### **10.5.3.1 Component Description**

The following paragraphs provides information that pertains to the major MCA components.

##### **10.5.3.1.1 Main Condenser**

The main condenser is located in the TB. The main condenser has two shells each mounted beneath and connected to a LP turbine by an expansion joint. The condenser is designed in accordance with the Heat Exchange Institute Standard for Steam Surface Condensers.

Vacuum breaker(s) are provided to break the vacuum and maintain atmospheric pressure within the condenser during shutdown. Connections are provided on each condenser shell to allow for the removal of air and other non-condensable gases.

The main condenser provides connections for the condensate pump minimum recirculation flow lines. The condenser hotwell receives makeup water from the CST. Excess water volume in the

condenser hotwell is drawn off by the condensate pumps and transferred to the CST. The hotwells of the two condenser shells are interconnected. The condensate pumps take suction from one of the hotwells.

Corrosion resistant alloys are used for condenser components exposed to steam or condensate where particularly severe duty is expected. The condenser tubes are titanium or stainless steel and the tubesheets are made of a compatible material.

Connections are provided to allow draining and cleaning of the condenser hotwell during plant shutdown.

The shielded area and controlled access for the main condenser is provided to protect personnel from radiation.

LP turbine exhaust steam, turbine bypass steam, and other miscellaneous power cycle drains, and vents are routed to the main condenser.

#### **10.5.3.1.2 Condenser Vacuum Pumps**

Two vacuum pumps are used as rapid evacuators to remove the air from the condenser shells and associated turbine, creating the initial vacuum which allows the startup of the plant. The vacuum pumps are used up to 5% reactor power before transfer to the SJAEs, due to the potential for H<sub>2</sub> concentration limits being reached.

The vacuum pumps can also run during shutdown when adequate steam is not available to operate the SJAEs.

A check valve is provided on each vacuum pump skid to prevent reverse flow, thereby preserving the condenser vacuum when the vacuum pump stops.

#### **10.5.3.1.3 Steam Jet Air Ejectors**

The SJAEs are sized in accordance with the "Venting Equipment Capacity" of Heat Exchange Institute's Standards for Steam Surface Condensers. Additionally, radiolytic oxygen and hydrogen entrained in the main condenser is removed by SJAEs.

The SJAEs function by using MS through the Turbine Auxiliary Steam System. The motive steam that condenses in the steam jet air ejector intercondenser flows through the loop seal to the condenser. The second stage air ejector is non-condensing. The air ejectors provide adequate motive force for the Offgas System (Chapter 11, Section 11.3.).

The steam supply to the second stage SJAEs ensures adequate dilution of hydrogen (less than 4% by volume) and prevent the non-condensable gases from reaching the flammable limit of hydrogen.

#### **10.5.3.1.4 Piping and Valves**

Any sections of pipe that may carry potentially explosive mixtures of hydrogen and oxygen are designed to contain the explosion.

All other piping is designed in accordance with ASME B31.1, Power Piping (Reference 10.5-2).

Valves exposed to higher hydrogen/oxygen mixture have bellows stem seals, double stem seals, or equivalent.

### **10.5.4 Materials**

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components,

corrodents and mitigating the degradation of materials specifically from corrosion (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

### **10.5.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.5-1 for system interfaces.

### **10.5.6 System and Equipment Operation**

The MCA is designed to operate during all modes of normal power plant operation and startup. The general operating logic for each of the operating modes is provided.

#### **10.5.6.1 Normal Operational Concept**

##### ***10.5.6.1.1 Normal Startup/Shutdown Mode***

The main condenser hotwell is filled from the CST, if drained, to the normal water level by automatic control of the normal and emergency makeup valves.

Prior to placing the condenser vacuum pumps in operation, the seal water separator is filled to the normal level and cooling water is flowing through the tube side of the seal water heat exchanger.

The condenser vacuum pumps are started to evacuate air from the condenser before the TGSS is placed into operation and the expansion joint seal water trough has been filled, if provided. The condenser vacuum pumps shut off automatically if abnormal levels of radioactivity are detected in the MSL and/or TB vent stack. Refer to Subsection 10.2.3 for information pertaining to TGSS.

Upon establishing a sufficient initial vacuum in the condenser, the condenser vacuum pumps remain in-service and hold the partial vacuum until motive steam is available for use by an SJAE skid. Steam flows from the NBS through the TASS to the SJAE.

Prior to placing an SJAE skid into operation, condensate is flowing through the tube side of the associated SJAE intercondenser.

The main condenser is also used to deaerate the reactor before criticality.

The Circulating Water System (CWS) is in operation prior to admitting any high energy fluids into the main condenser. Refer to Section 10.8 for information related to the CWS.

The condenser hotwell and its level control system are designed to accommodate the reactor vessel inventory shrink and swell associated with pressurization/de-pressurization during normal startups and shutdowns.

The main condenser acts as a heat sink in the initial phase of reactor cooldown during a normal plant shutdown.

Condenser vacuum breaker(s) are opened, slowing the turbine during shutdown.

##### ***10.5.6.1.2 Normal Operation***

The main condenser condenses the LP turbine exhaust steam, stores condensate, acts as a condensate surge volume, and supplies condensate to the condensate pumps. The main condenser also accepts vents and drains discharge from various points in the cycle.

SJAE units removes air in-leakage and radiolytic gases from the condenser shells, and in conjunction with the CWS, maintains a vacuum in the condenser shells.

### **10.5.6.2 Off-Normal Operational Concept**

#### **10.5.6.2.1 Transient Operation**

The main condenser hotwell acts as surge volume in the following events:

1. Tube failure of FW heaters, SJAE intercondensers, MSRs, and other miscellaneous heat exchangers.
2. High-water level in the FW heaters and reheater drain tanks
3. Relief valve discharge for system overpressure protection as well as condensate from the MSLs through their drain subsystem

The condenser accepts steam and drains from various points in the turbine cycle, including those that require steam dumps during turbine generator load rejection.

The condenser hotwell receives overboard flow from the Shutdown Cooling System (SDC) (described in Chapter 9, Subsection 9A.2.3) during reactor heatup. The CUW can overboard to the LWM Chapter 11, Section 11.2) or MCA depending on radiation monitoring. The primary flow path is to MCA with the option for diverting to LWM.

#### **10.5.6.2.2 Load Rejection**

The main condenser accepts up to 25% rated steam flow via the TBS (Chapter 10, Section 10.7) in the event of a turbine trip or load reject.

There is enough condensate stored in the hotwell to allow operation of the condensate pump(s) during this transient.

#### **10.5.6.2.3 High Condenser Backpressure**

The pressure in the condenser may rise during normal operation due to low circulating water flow, high circulating water temperature, or high condenser duty. In any case, as condenser pressure increases above normal levels, an alarm annunciates. A further increase in pressure to the turbine trip limit results in a turbine trip. As pressure increases further beyond an established condenser limit, the turbine steam bypass valves are inhibited from opening and if open, they close to prevent overpressurization of the condenser shell. If the pressure continues to increase, the MSCIVs close.

The SJAE skid on standby may be brought online if a high absolute pressure exists in the condenser.

#### **10.5.6.2.4 Loss-of-Offsite Power**

The main condenser accommodates the turbine bypass flow for at least six seconds following a trip of the circulating water pumps due to a loss of preferred power without exceeding the TBV isolation pressure setpoint in the condenser.

### **10.5.7 Instrumentation and Control**

The MCA provides the instrumentation to control and monitor (indicate and alarm) system operation. Package (skid) instrumentation is provided to verify vacuum pumps and steam jet air ejector performance in addition to normal operation and control. Instrumentation is included and installed to provide an accurate plant heat balance.

#### **10.5.7.1 Instrumentation**

Major system parameters, i.e., hotwell levels; and process flow rate, pressures, and temperatures are indicated and alarmed as required to provide operational information and performance assessment.

#### **10.5.7.2 Controls**

The MCA is operated and controlled from the MCR. Major system control functions, i.e., hotwell level, SJAE performance, and valve positioning logics, are based on system operating parameters.

#### **10.5.8 Monitoring, Inspection, Testing and Maintenance**

The equipment and components of the MCA are designed for easy inspection and maintenance with the intent to reduce or eliminate radiation exposure potential. Routine testing of the MCA is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

#### **10.5.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.5.10 Performance and Safety Evaluation**

The MCA is capable of performing its Safety Category normal functions as presented in Subsection 10.5.1. The MCA does not perform any Safety Category off-normal functions.

The MCA is capable of measuring main condenser vacuum (pressure) to meet Safety Category 2 and Safety Category 3 requirements. The system also provides a heat sink, removes non-condensable gases from the condenser shell and supplies condensate to meet Safety Category 3 requirements listed in Subsection 10.5.2.

#### **10.5.11 References**

- 10.5-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.5-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.5-1: Main Condenser and Auxiliaries System Interfaces**

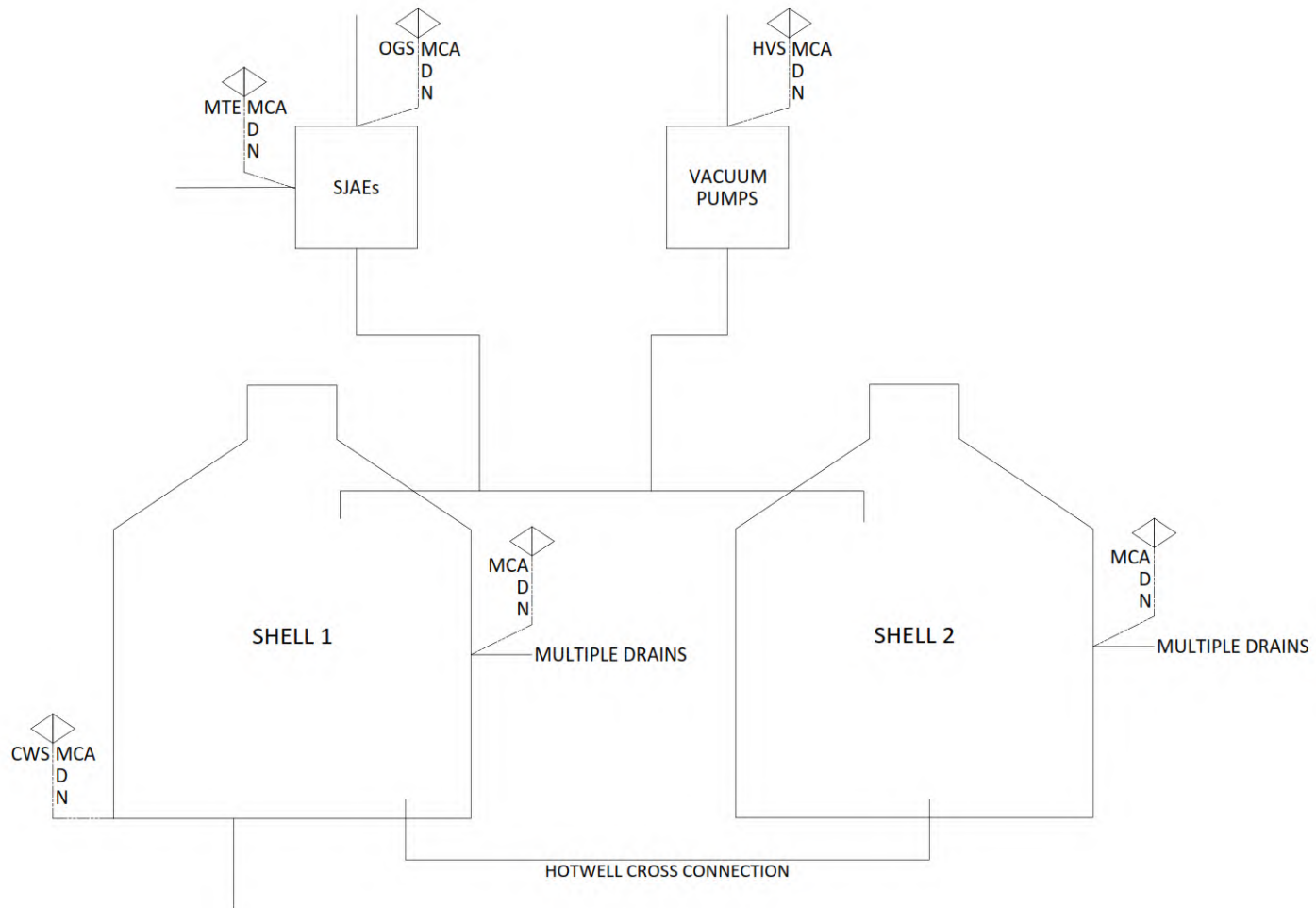
<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	The NBS steam line drains any condensate to the main condenser	At the main condenser
SC2 and SC3 Instrumentation and Controls	Main condenser SC2 and SC3 instrumentation is used for controls	At MCA equipment
Non-Safety Instrumentation and Controls	Main condenser Non-Safety instrumentation is used for controls	At MCA equipment
Control Panel System	Main condenser operation status alarms and indications	At MCA equipment
Process Sampling	Main condenser hotwell sampling	At the main condenser
CRD System	Drain for CRD pump minimum flow line return to MCA	At the main condenser
Shutdown Cooling System	The main condenser receives overboard flow from the SDC	At the main condenser
Reactor Water Cleanup System	The main condenser receives overboard flow from the CUW	At the main condenser
Liquid Waste Management System	Makeup water to the main condenser via the CST	At the main condenser
Offgas System	SJAE discharge to the Offgas System	At the SJAE exhaust
Main Turbine Equipment	The LP turbine exhaust steam discharges to the main condenser. The TASS provides MS to the SJAEs	At the main condenser At the SJAEs
Condensate and Feedwater Heating System	The main condenser hotwell supplies condensate to the condensate pumps. The main condenser receives the pump recirculation flow, drains, and vents from the CFS system	At the main condenser
Moisture Separator Reheater System	The main condenser receives drains and gases from the MSR System	At the main condenser
Turbine Bypass System	The main condenser receives turbine bypass steam	At the main condenser



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Circulating Water System	Cooling water to the main condenser	At the main condenser
Plant Cooling Water	Cooling water for condenser vacuum pumps	At the condenser vacuum pumps
Plant Pneumatics Equipment	Provides dry compressed air for valves and instrumentation	At MCA equipment
Non-Safety Electrical Distribution System	Provides SCN low voltage power	At MCA equipment
Heating Ventilating and Cooling System	The Heating Ventilating and Cooling System will receive gaseous discharge from the condenser vacuum pump separators	At the condenser vacuum pumps
Equipment and Floor Drain System	Receives drains from the main condenser hotwell and the condenser vacuum pumps	At MCA equipment
Water Gas and Chemical Pads	Provides fill and makeup water to the condenser vacuum pumps	At MCA equipment

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 10.5-1: Main Condenser and Auxiliaries System**

## **10.6 Moisture Separator Reheater**

The MSR is composed of one moisture separator reheater, three drain tanks and the piping from the HP turbine exhaust (cold reheat) to the MSR, from the MSR to LP turbines (hot reheat) inlet, piping up to the intermediate stop valves and intercept valves as well as piping and valves from MSL equalizing header for 2nd stage reheating. The MSR has one stage of moisture separation and two stages of reheating.

### **10.6.1 System and Equipment Functions**

The MSR System performs the following functions during normal and off-normal conditions.

#### **10.6.1.1 Normal Functions (Non-Safety Category)**

The MSR System performs a Non-Safety Category normal function to reheat HP turbine exhaust prior to use by the LP turbines (TBD).

#### **10.6.1.2 Normal Functions (Safety Category)**

The MSR System provides the following Safety Category functions during normal conditions:

1. The MSR System repositions level control valves, for both normal and alternative flow paths, based on signals from I&C systems, to drain condensed steam from the moisture separator reheater drain tanks.
2. The MSR System provides hot condensate for FW heating.
3. The MSR System provides indication of Moisture Separator Reheater Drain Tank water level.

#### **10.6.1.3 Off-Normal Functions (Non-Safety Category)**

The MSR System does not provide any Non-Safety Category functions during off-normal conditions.

#### **10.6.1.4 Off-Normal Functions (Safety Category)**

The MSR System does not provide any Safety Category functions during off-normal conditions.

### **10.6.2 Safety Design Bases**

The MSR is categorized as Non-Nuclear Seismic. The MSR routes hot condensate from the MSR to FW heaters numbered 3, 4, and 5 as a Safety Category function as well as provides indication of Moisture Separator Reheater Drain Tank water level. Additionally, the system provides an alternate drain path from the MSR drain tanks to the condenser. These Safety Category functions serve to maintain reactor parameters within normal ranges during normal operations.

The design of the MSR meets CNSC requirements specified in REGDOC-2.5.2 Section 6.1 (Reference 10.6-1) as it relates to design for Defence-in-Depth and Section 7.1 as it relates to classification of SSC.

### **10.6.3 Description**

Figure 10.6-1 depicts the MSR. The MSR is located in the TB. The MSR reheats the HP turbine exhaust and then directs the reheated steam into the LP turbine. The purpose of this system is to dry and reheat the expanded steam from the HP turbine exhaust (cold reheat steam) to improve the cycle efficiency and reduce liquid impingement erosion and FAC in the LP turbines.

The MSR is protected by a safety relief valve that ensure that it is not overpressurized under the most adverse operating conditions associated with the turbine generator operations. The relief valve is located on the shell of the MSR with an outlet discharging to the condenser.

Steam from the HP turbine exhaust (cold reheat steam) enters the bottom of the MSR where steam separators remove entrained moisture before passing the steam to the reheater sections. Heating steam to the first stage of reheat in the MSR is supplied by the Extraction Steam Subsystem. Refer to Subsections 10.2.3 and 10.2.3.1.5 for additional information pertaining to the Extraction Steam Subsystem. The second stage reheating steam to the MSR is supplied by the MS subsystem (Section 10.4). Hot reheat steam is supplied to the two LP turbines through the reheat intermediate stop and intercept valves, which are located upstream of the LP turbines inlet nozzles.

Removed moisture is drained to the drain tank. Steam condensed from the reheater stages of the MSR is also drained to drain tanks. FAC resistant materials are used for all components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of FAC resistance of the material is consistent with the temperature, moisture content and velocity of the wet steam to which the component is exposed.

Provisions are made to evacuate and purge all reheater tube bundles prior to pressurization to minimize distortion when heating steam is admitted to the tube bundle.

Drains are provided at the low points in the cold and hot reheat piping. These drains are placed as close to the turbine as possible and are designed in accordance with ASME TDP-2 (Reference 10.6-2). Two valves are provided in series in each drain line. The upstream valve is air-operated (fail open). This drain valve is located as close as possible to the drain pot to minimize the amount of water trapped upstream from the closed valve. The second valve is manually operated.

A power operated non-return valve and control valve is installed in the supply piping to the first stage reheater.

A control valve is installed in the supply piping to the second stage reheater tube bundle. A control valve is installed upstream of this valve and a manual valve downstream. Piping containing a control valve is installed to bypass the control valve station.

All horizontal runs of the cold reheat steam, hot reheat steam, and first and second stage reheating steam is sloped toward the MSR for drainage where appropriate.

The routing of the small-bore miscellaneous drain piping is designed to slope downward in the direction of the flow.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSC.

### **10.6.3.1 Component Description**

The following paragraphs describe the major components of the MSR.

#### **10.6.3.1.1 Moisture Separator Reheaters**

The MSR has one stage of moisture separation and two stages of reheating.

The MSR has cylindrical, horizontal, combined moisture separator shells with steam separators and closed U-tube reheater tube bundles. Steam separators are made of stainless steel and are adequately supported to resist aerodynamic, fluid elastic or any other type of vibration. Analyses supported by testing is performed to ensure good flow distribution to the inlet face of the moisture separator to maximize effectiveness and to eliminate the possibility of structural damage from locally high steam velocities.

The MSR is designed in accordance with ASME BPVC, Section VIII, Division 1, (Reference 10.6-3) Unfired Pressure Vessels, and Thermal Exchanger Manufacturers Association Standards.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

The reheater tube supports are designed and spaced to prevent wear or damage due to aerodynamic, fluid elastic or any other type of vibration of the reheater tubing.

Layup provisions which are appropriate for the MSR materials are provided. Tube side layup is performed with treated water, nitrogen, or dry clean air according to paragraph 304 of ASME NQA-1 (Reference 10.6-4). Shell side layup is performed with treated water or dry clean air.

The MSR is protected by a safety relief valve to prevent overpressurization.

The relief valve is located on the shell of the MSR with outlets discharging to the condenser.

#### **10.6.3.1.2 MSR Drain Tanks**

The MSR includes three drain tanks; one for the moisture separation stage, one for the first stage of reheat, and one for the second stage of reheat.

The drain tanks are designed in accordance with ASME BPVC, Section VIII, Division 1.

Moisture separator drain tank piping is designed to handle two-phase flow.

#### **10.6.3.1.3 Piping and Valves**

MSR piping is designed to the requirements of ASME B31.1, Power Piping (Reference 10.6-3) as applicable.

A power operated non-return valve and control valve is located in the supply piping to first stage reheater tube bundle for isolation purposes.

An air-operated control valve is located in the MS supply piping to each second stage reheater tube bundle to modulate the steam flow. A control valve is located upstream of this valve and a manual valve is located downstream for isolation purposes. Piping containing a control valve is provided to bypass the control valve in the event it is unavailable.

Steam shutoff, and control valves have both automatic and manual control to regulate the second stage reheater temperature. Manual control includes the capability to manually open and close the shut-off valve and position the control valve.

#### **10.6.3.1.4 Drains**

Each moisture separator and reheater tube bundle has its own separate drain system.

Moisture separator drains are sized for self-venting gravity downflow into the moisture separator drain tanks to allow the system to vent properly during plant operating conditions. In addition, a separate vent is taken from above the water level in each drain tank back to the moisture separator to avoid any possibility of lock-up during plant operating conditions.

#### **10.6.3.1.5 Vents**

Venting and/or other arrangements are provided to ensure the continuous removal of non-condensable gases from the reheater tube bundle and also ensure the steady removal of condensate from the reheater tubes with minimum subcooling. Thermal-hydraulic instabilities in the reheater tube bundle is minimized to such an extent no short or long-term operational or reliability problems occur. Vent piping has no low points and is sloped to ensure adequate drainage.

### **10.6.4 Materials**

Material and equipment selection for the MSR is based on the range of normal environmental conditions that may exist at the location of the system equipment and components.

FAC resistant materials is used for components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of FAC resistance of the material is consistent with the temperature, moisture content and velocity of the wet steam to which the component is exposed.

#### **10.6.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.6-1 for MSR interfaces.

#### **10.6.6 System and Equipment Operation**

The MSR is designed to operate during modes of normal power plant operation, including startup and shutdown. The general operating logic for each of the operating modes is provided below.

##### ***10.6.6.1.1 Normal Operational Concept***

The MSR receives steam from the HP turbine exhaust. Moisture is removed and the steam is reheated for use in the LP turbine.

HP turbine extraction steam is directed through open isolation valves to the first stage reheater.

MS is directed through open control valves to the second stage reheater. The second stage heating medium is HP steam supplied by the MS subsystem as discussed in Subsection 10.6.3. This steam supply is regulated by flow control valves to control the temperature of the steam at the outlet of the MSRs.

##### **10.6.6.2 Off-Normal Operational Concept**

###### ***10.6.6.2.1 Turbine Trip***

Upon a turbine trip, the first stage reheater isolation valve and the second stage reheater control valve close, and the cold reheat drains open. Hot reheat steam is isolated by closing the intermediate stop valve and intercept valves. The MSR is isolated and condensate is drained from the MSR.

###### ***10.6.6.2.2 Load Following***

The MSR accommodates load following with CRD operation between 100% and 50% of rated power with ramp rates up to  $\pm 0.5\%$  per minute. The design accommodates approximately 20,805 equivalent daily load following cycles. Refer to Subsection 10.2.2 for information pertaining to load following.

###### ***10.6.6.2.3 System Startup***

During startup, the main turbine stop and control valves, first stage reheater isolation valve, and second stage reheater control valve are closed. As the main turbine stop and control valves begin to open allowing steam to be admitted to the turbine, cold reheat steam enters the MSR. The first stage reheater isolation valve is opened allowing HP extraction steam to enter the first stage reheater. The second stage reheater control valve is slowly opened to increase the hot reheat temperature at which point the second stage reheater control valve is fully open.

###### ***10.6.6.2.4 Turbine Shutdown***

The main turbine stop and control valves remain open and the MSR continues to receive cold reheat steam from the HP exhaust and supply hot reheat steam to the LP turbine until the main turbine falls below the manufacturer's specified kW output. At this point, the main turbine stop and control valves, the first stage reheater isolation valves, and the second stage reheater control valves are closed.

### **10.6.7 Instrumentation and Control**

Instruments are designed to withstand or be protected from transient pressures and flow resulting from the transitions between the various modes of system operation.

All instruments, excluding welded-in flow nozzles, are removable for servicing without cutting pipes or grinding welds.

Instrument scale, size and style is selected in compliance with Human Factors Engineering requirements (Chapter 18). The scale graduations are consistent with the required accuracy and range of the measurement.

#### **10.6.7.1 Special Features**

The MSR loop contains sufficient instrumentation to:

- Monitor and maintain appropriate heating steam flow for proper reheater operation
- Monitor and operate drain pot drain valves
- Verify reheater heat transfer quality
- Initiate turbine trip upon two-out-of-three coincident logic for high-water level in the moisture separator section of the MSR (Turbine Water Induction Prevention requirement)
- Provide necessary status indication of key valves and annunciation of abnormal conditions during operation

#### **10.6.7.2 Instrumentation**

The instrument features of the MSR System are as follows:

1. The moisture separator section is equipped with triple redundant level instruments mounted on cross connected independently valved bridles. The instruments initiate turbine trip based on high moisture separator water level.
2. Both the cold reheat steam and hot reheat steam piping is equipped with triple redundant pressure instrumentation and redundant temperature instrumentation.
3. The steam supply piping to the reheaters is also equipped with pressure and temperature instrumentation, and flow instrumentation to detect tube leaks.
4. MSR is equipped with differential pressure transmitters to measure the pressure drop across the MSR.
5. Each cold reheat and hot reheat drain line is equipped with temperature instrumentation.

The MSR includes the sensors recommended in ASME PTC 12.4.

Refer to Chapter 7 for information pertaining to triple redundancy.

#### **10.6.7.3 Control**

Key control features of the MSR are included below. This system is automatically controlled.

Manual initiation and shutdown of the System is provided from the MCR.

Cold reheat drain valves manually open and close from the MCR. The cold reheat drain valves are fail open valves that auto open when level instruments in their respective drain pots give a high or high-high level signal, turbine trip, load rejection, or drain group open signal and auto close on drain group close signal.

Hot reheat drain pot air-operated drain valves manually open and close from the MCR. The hot reheat drain valves are fail open valves that auto open when level instruments in their respective drain pots give a high or high-high level signal and auto close on drain group close signal.

The control valve in the supply piping to the first stage reheater tube bundle automatically and manually opens and closes from the MCR. No temperature control is carried out for first stage heating steam. Flow rate and steam conditions are dependent upon the heat and mass balance of the HP turbine.

The control valve located in the MS supply piping to the second stage reheater tube bundle automatically and manually opens and closes from the MCR and is designed to fail closed. The NBS control valve upstream of the MSR control valve is designed to open manually from the MCR.

The following primary MSR displays, and alarms are provided:

- Main Control Console Displays
  - Heating steam flow, pressure, and temperature
  - Reheat steam pressure and temperature
  - Valve positions
- MCR Alarms
  - Drain pot high or high-high level
  - High level in the moisture separator section

#### **10.6.8 Monitoring, Inspection, Testing, and Maintenance**

Routine testing of the MSR is conducted in accordance with plant requirements for demonstrating system and component operability and integrity. Routine testing includes inspection for liquid impingement erosion and heat exchanger tube integrity inspections per applicable codes and standards.

The equipment and components of the MSR are designed for easy inspection and maintenance during shutdown/refueling.

The MSR is directly accessible by the TB crane. Lift capability and space is provided to permit removal and replacement of an entire MSR as well as tube bundles without requiring the removal of any other major component or large size piping.

Tube bundle replacement does not require removal of permanent walls, major piping, or cables trays. Adequate space is provided for laydown of replacement MSR tube bundles and associated equipment. MSR tube bundle pull fixtures are specifically designed for each location.

Flow restriction orifices are located to allow ease of access for replacement.

#### **10.6.9 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.6.10 Performance and Safety Evaluation**

The MSR is designed to operate during modes of normal power plant operation, including startup and shutdown.

The Safety Category functions associated with the MSR provide:



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

- A route for hot condensate from the MSR to FW Heaters No. 3, 4, and 5
- An alternative drain path from the MSR drain tanks to the condenser
- Indication of MSR Drain Tank water level

These Safety Category functions serve to maintain reactor parameters within normal ranges during normal operations.

**10.6.11 References**

- 10.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.6-2 ASME TDP-2, "Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants," American Society of Mechanical Engineers.
- 10.6-3 ASME BPVC-VIII, "BPVC Section VIII-Rules for Construction of Pressure Vessels Division 1," American Society of Mechanical Engineers.
- 10.6-4 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 10.6-5 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

**Table 10.6-1: Moisture Separator Reheater System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Nuclear Boiler System	Provides steam for second MSR reheat stage	MSR 2nd stage heating
SC3 Instrumentation and Distributed Controls and Information System	Turbine Controls System MSR SC2 and SC3 I&C System SC3 DCIS controls and the MSR valves	Control for MSR System valves and drains MSR Shell Level transmitters with setpoint "Initiate Turbine Trip on High Level"
Process Radiation and Environmental Monitoring System	Samples taken continuously to monitor water quality	Drain Tank Drain Lines
Condensate and Feedwater Heating System	MSR Drain tanks drain into FW heaters shells	FW heaters shell inlets
Main Turbine Equipment	Provides HP turbine exhaust to be reheated and extraction steam for heating Discharges reheated steam to LP turbine	HP Turbine Exhaust HP Extraction Reheat steam intermediate stop and intercept valves
Main Condenser and Auxiliaries	Relief valve, MSR Drain Tank, 1 <sup>st</sup> Stage Reheater Drain Tank, and 2 <sup>nd</sup> Stage Reheater Drain Tank, and drain pots drain into the condenser	Relief valve, MSR Drain Tank, 1 <sup>st</sup> Stage Reheater Drain Tank, and 2 <sup>nd</sup> Stage Reheater Drain Tank, and drain pots discharge piping
Plant Pneumatics System	Air supply for AOVs	AOV
Non-Safety Electrical Distribution System	Power for MOVs	MOV
Turbine Building Structure	Accommodate loads for the MSR System	Structure
Equipment and Floor Drain System	Drainage for the system during Maintenance	Local Drains

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

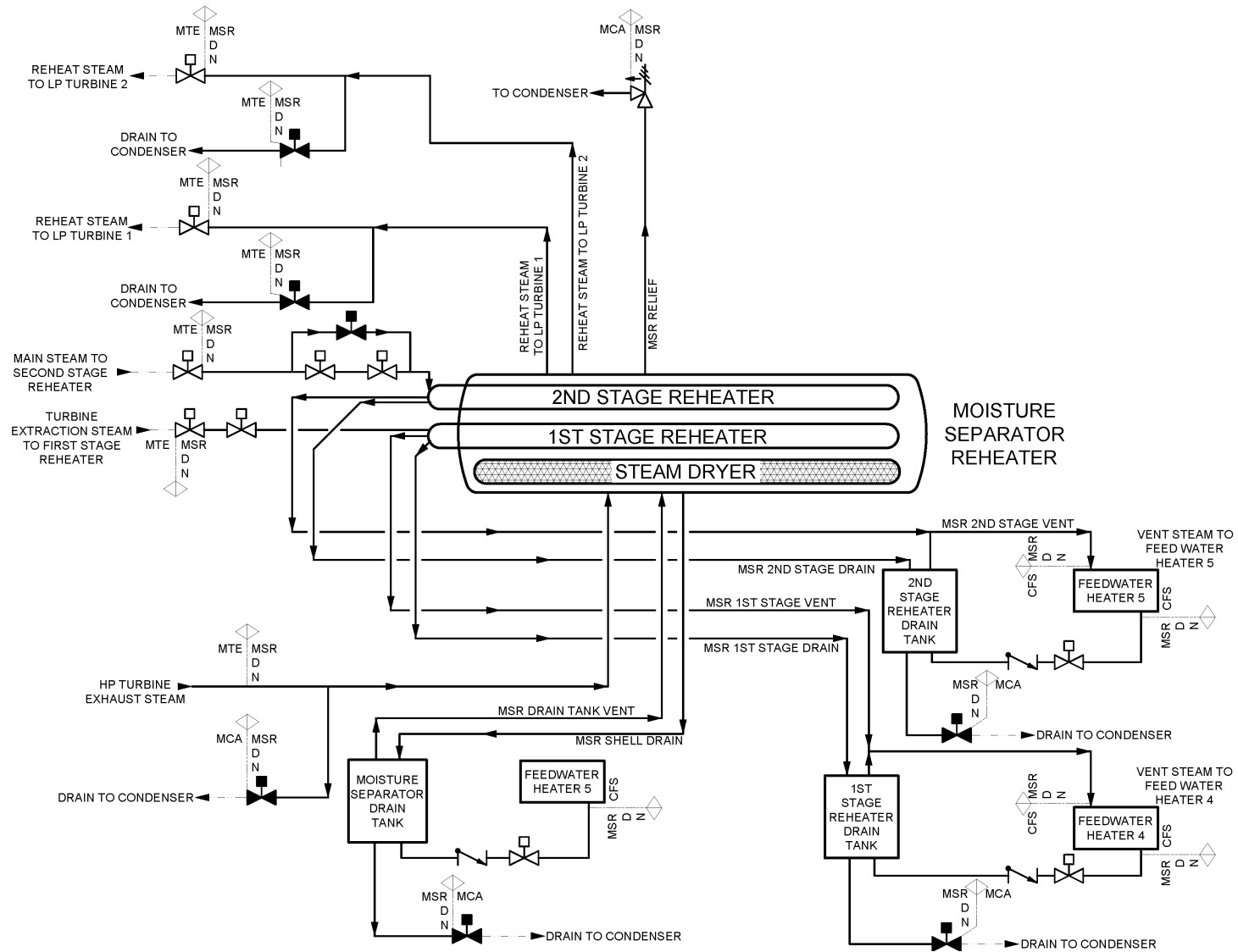


Figure 10.6-1: Moisture Separator Reheater System

## **10.7 Turbine Auxiliary Systems**

This section describes the TBS. The TBS takes excess steam from the NBS (Chapter 5), expands the steam to lower pressure, then uses spargers to deliver the steam to the condenser (Subsection 10.5.1).

### **10.7.1 System and Equipment Functions**

#### **10.7.1.1 Normal Functions (Non-Safety Category)**

The TBS does not perform any Non-Safety Category functions during normal conditions.

#### **10.7.1.2 Normal Functions (Safety Category)**

The TBS performs the following Safety Category functions during normal conditions:

- Reposition the TBVs to control reactor dome pressure in Modes 1 (Power Operation) and 2 (Startup)
- Remove decay heat from the reactor and discharge to the main condenser in Modes 3 (Hot Shutdown) and 4 (Stable Shutdown)

#### **10.7.1.3 Off-Normal Functions (Non-Safety Category)**

The TBS does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.7.1.4 Off-Normal Functions (Safety Category)**

The TBS has three Safety Category functions during off-normal conditions:

- Control reactor dome pressure using the TBVs based on signals from I&C systems in Modes 1 (Power Operation) and 2 (Startup).
- TBVs will fast open in response to a turbine trip when demanded by I&C systems in Mode 1 (Power Operation).
- TBVs will close on high main condenser pressure when demanded by the I&C system in Modes 1 (Power Operation) and 2 (Startup).

### **10.7.2 Safety Design Bases**

The TBS is designed to operate in modes of normal power plant operation and startup. The TBVs open as required to maintain reactor pressure following a load reduction or load rejection or whenever the turbine is not able to flow the steam demanded by the RPC. As part of DL2 the TBVs receive a signal from the SC2 and SC3 I&Cs System to open and close when signaled to do so.

The design of the TBS meets CNSC requirements specified in REGDOC-2.5.2 Section 6.2 (Reference 10.7-1) as it relates to maintaining the plant in a normal operational state.

### **10.7.3 Description**

Figure 10.7-1 depicts the TBS. The TBS is located in the TB and is comprised of the following:

1. Two TBVs
2. TBV hydraulic actuators, hydraulic accumulators, supply and drain piping, servo valves, solenoid valves, check valves, and position transmitters
3. Piping, fittings, instruments, and valves from the inlet connections of the TBV to the sparger connections at the main condenser

Two TBVs receive steam from the NBS piping connected to equalization header and discharge the steam to the main condenser.

Threaded and flanged connections are not used except where required to permit component removal for maintenance.

The TBVs have accumulators, which allow for limited operation of TBV upon electrical power failure. In the event power is lost to the hydraulic pumps but is still available to the control system, the hydraulic actuation system is capable of either cycling the valves open and closed three times or opening the valves for a minimum time of six seconds and then re-closing them under the direction of the RPC. In the event electrical power is lost to the control system (regardless of whether the hydraulic unit has power or not), the valves automatically close and remain closed.

The TBVs have sufficient capacity and transient response capability to permit generator synchronization with the power grid without a pressure transient resulting in an excessive power excursion or reactor water level perturbations.

The TBV has sufficient capacity and transient response capability to permit stable operation of the automatic pressure control of the reactor and to permit controlled cooldown of the plant to the point where the SDC can be placed in service.

The total rated turbine bypass capacity is 25% of the nuclear boiler rated steam flow at rated turbine inlet pressure.

The TBS is designed such that no failure of a single component can disable more than 50% of the installed bypass capacity

#### **10.7.3.1 Component Description**

##### **10.7.3.1.1 Turbine Bypass Valves**

Two (2) TBVs are provided. Each TBV is a control valve operated by a hydraulic actuator. The actuator is equipped with a spring which is capable of shutting the valve against the design steam pressure. The TBVs are capable of modulating steam flow to meet system requirements.

Individual TBVs are designed to fail closed upon loss of control power or loss of hydraulic pressure to their operators. Individual TBV hydraulic accumulators have the capability to stroke the valves with HPU failure and meet opening time and duration requirements. The accumulators in combination with hydraulic check valves ensure that no single hydraulic failure disables more than 50% of the TBVs. The accumulator hydraulic supply is also designed to ensure that upon LOOP or station auxiliary power, TBVs can still be opened for a minimum time of six (6) seconds and re-close.

American National Standards Institute (ANSI FCI 70-2) Class V leakage class is specified for bypass valves to minimize wire drawing and other damage to the valve caused by leakage across the seat.

##### **10.7.3.1.2 Turbine Bypass System Piping**

TBS piping is designed to ASME B31.1, Non-Nuclear Seismic requirements (Reference 10.7-2) as applicable.

Piping supports are designed for water-filled line loads under static loading conditions that may be encountered in plant operations. In addition, provisions are made for conveniently supporting (e.g., by pinning hangers) the dead weight loads imposed during hydrostatic testing.

The MS piping from the equalization header slopes upward to the TBV inlets and the discharge piping downstream of the TBVs slopes downward to the main condenser with a slope of at least

1/100 of run with no low points. Suitable low point drains are provided by the NBS to prevent excessive moisture from accumulating upstream of the TBVs.

Spargers distribute the steam within the main condenser as required.

#### **10.7.4 Materials**

Material and equipment selection for the TBS is based on the range of normal environmental conditions that exist at the location of the system's equipment and components.

#### **10.7.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.7-1 for TBS interfaces.

#### **10.7.6 System and Equipment Operation**

The TBS is designed to operate in all modes of normal power plant operation and startup.

##### **10.7.6.1 Normal Operational Concept**

###### **10.7.6.1.1 Startup Mode**

During startup, the turbine stop and control valves are initially closed. The RPC opens the TBVs as required, bypassing steam to the main condenser, to control reactor pressure allowing the reactor to operate to a rated condition and to minimize pressure transients in the reactor during initial steam admission to the turbine. As steam is admitted to the turbine, the TBVs close as required to maintain reactor pressure.

###### **10.7.6.1.2 Normal Operation**

During normal operation, the TBVs are maintained in the closed position.

The TBVs open as required to maintain reactor pressure following a load reduction or load rejection. The TBS is capable of accommodating a <25% load rejection with the TBVs modulating as required to maintain reactor pressure and support operation of the turbine under reduced load.

###### **10.7.6.1.3 System Shutdown**

The TBVs initially open upon turbine trip, then modulate as required to maintain reactor pressure. During shutdown, reactor power is reduced by rod insertion at specified rates and sequences. The turbine stop and control valves remain open until the main turbine power falls below the manufacturer's specified kW output limit. At this point, the turbine stop and control valves close, and the TBVs open and modulate as required to control steam pressure.

#### **10.7.7 Instrumentation and Control**

The TBVs open in response to redundant bypass demand signals received from the SC2 and SC3 I&C System (Chapter 7) whenever the actual steam pressure exceeds the preset steam pressure by a preset amount. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. The number of TBVs required to open for bypass operation is determined by the SC2 and SC3 I&C System, and the TBVs modulate as required in response to changes in the bypass demand, discharging steam to the main condenser.

The following signals are monitored in the MCR:

- Temperature in each bypass line for bypass steam leakage detection
- Valve position indication for each TBV
- TBV operable status and annunciation of any signals that cause the valves to become inoperable

Three SC3 valve open/close sensors are provided by the SC2 and SC3 I&C System and installed at each TBV detects the 10% open position and provides valve status to the SC2 and SC3 I&C System reactor protection logic. These position sensors are fail-safe such that they cannot prevent actuation of the reactor protection function.

SC3 valve position transmitters are provided and installed at each TBV. These valve position transmitter signals are used to perform input voting as applicable and are used for the reactor pressure controller of the SC2 and SC3 I&C System DCIS. The TBV operational status are used for monitoring functions in SCN Non-Safety I&C System DCIS.

Temperature elements installed in each TBV discharge pipe near the main condenser inlet provides temperature signals to the RPC for bypass steam leakage detection.

Triply redundant bypass valve position transmitters installed at each TBV provide valve position signals to the RPC.

Control and monitoring of the TBS is accomplished by the RPC and the logic contained therein.

Triply redundant servo valves and fast-acting solenoid valves provide hydraulic actuation of the TBVs upon receiving demand signals from the RPC.

The RPC determines bypass operating conditions and setpoints which require fast opening of the TBVs and provides the corresponding demand signals to the TBS.

Automatic and manual control of the TBS is provided. TBV controls, including manual initiation and shutdown, are located in the MCR to facilitate system operation.

#### **10.7.8 Monitoring, Inspection, Testing, and Maintenance**

The equipment and components of the TBV are designed for ease of inspection and maintenance. The TBVs are oriented to facilitate maintenance activities. Routine testing of the TBV is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. Periodic inspections are performed on a rotating basis within a preventive maintenance program in accordance with manufacturer's recommendations. Testing is in accordance with ASME OM Code.

#### **10.7.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.7.10 Performance and Safety Evaluation**

The TBS is designed to operate in all modes of normal power plant operation and startup. As part of DL2, the TBVs receive a signal from the RPC to open and close when signaled to do so.

1. The TBVs are capable of modulating steam flow to meet system requirements.
2. The TBVs are designed to fail closed upon loss of control power or loss of hydraulic pressure to their operators.
3. The TBV hydraulic accumulators have the capability to stroke the valves with hydraulic power and meet opening time and duration requirements after loss of power to the EHC pumps.
4. The accumulators in combination with hydraulic check valves ensure that no single hydraulic failure disables more than 50% of the TBVs.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

5. The accumulator hydraulic supply is also designed to ensure that upon LOOP or station auxiliary power, TBVs can still be opened for the minimum time and re-closed.

**10.7.11 References**

- 10.7.1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.7-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.



**Table 10.7-1: Turbine Bypass System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System	The NBS provides steam to TBS when bypass is necessary	TBVs
SC2 and SC3 Instrumentation and Controls	SC3 valve position transmitters on each TBV. Control of TBVs	Turbine Bypass Valve
Non-Safety Instrumentation and Controls	Temperature Sensors for leakage detection and TBV performance monitor	Temperature instruments
Instrument Racks System	Instruments mounted on racks outside the TB shielded area for cycle access	Temperature instrumentation
Main Turbine Equipment	Seal Steam from TGCS Hydraulic Fluid from EHC	Turbine Bypass Valve
Main Condenser and Auxiliaries	Sparger into condenser	Condenser Sparger Nozzle

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

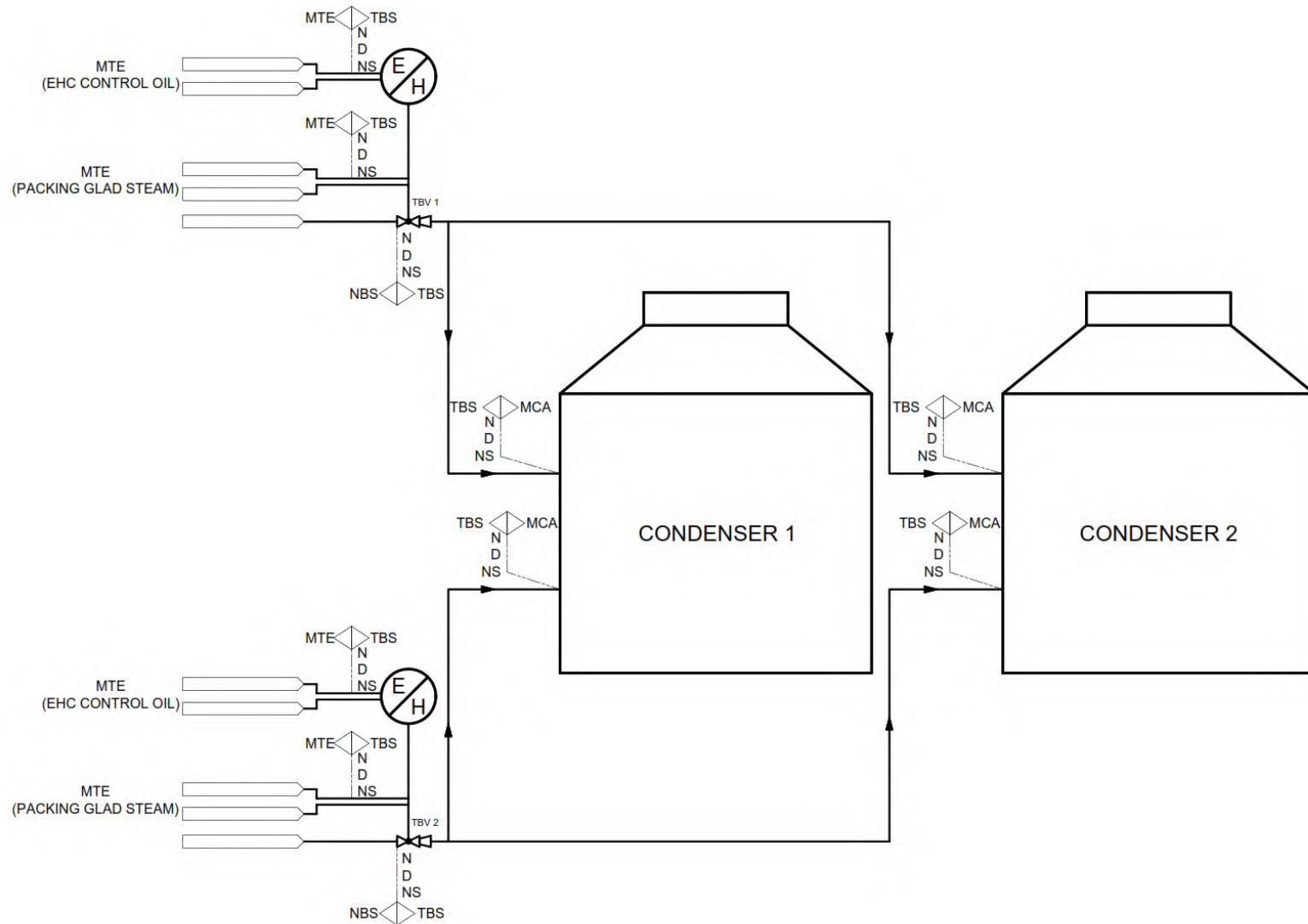


Figure 10.7-1: Turbine Bypass System Interfaces

## **10.8 Circulating Water System**

The CWS has two subsystems, the main condenser supply and the PCW supply. The main condenser supply uses 2x50% pumps to provide cooling water to the MCA (Subsection 10.5.1) during all modes of condenser heat removal. The plant cooling water supply uses 2x100% pumps to provide cooling water to the PCW (Chapter 9, Subsection 9A.2.1) heat exchangers for all normal and abnormal operating modes. All CWS pumps are located in bays in the intake structure of the Normal Heat Sink (NHS) (Chapter 9, Subsection 9A.2.5).

### **10.8.1 System and Equipment Functions**

The CWS performs the functions listed below during normal and off-normal conditions.

#### **10.8.1.1 Normal Functions (Non-Safety Category)**

The CWS performs the following Non-Safety Category functions during normal conditions:

- Removes accumulated air and other gases from the water boxes through water box vents
- Provides the means to drain the condenser supply and return water boxes

#### **10.8.1.2 Normal Functions (Safety Category)**

The CWS performs the following Safety Category functions during normal conditions:

- Provides means to reject heat from the MCA to the environment through the NHS
- Provides means to reject heat from the PCW heat exchangers to the environment through the NHS

#### **10.8.1.3 Off-Normal Functions (Non-Safety Category)**

The system does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.8.1.4 Off-Normal Functions (Safety Category)**

The CWS continues to provide its Safety Category functions during off-normal conditions. Upon a Loss-of-Offsite Power (LOOP), the NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW. The two PCW supply pumps are powered from the diesel generators via the SC3 Electrical Distribution System, during a LOOP.

### **10.8.2 Safety Design Bases**

The safety design bases for the CWS are included below. Codes and standards related to pressure boundary are identified in the component descriptions in Subsection 10.8.3.1 as applicable. CWS system equipment is categorized as Non-Nuclear Seismic.

SC3 Requirements:

- As part of DL2, the CWS has an integral support function to cool PCW.
- The CWS provides a means to reject heat from the MCA to the environment through the NHS.

The design of the CWS meets CNSC requirements specified in REGDOC-2.5.2 Section 8.7 (Reference 10.8-1) as it relates to heat transfer to an ultimate heat sink.

### **10.8.3 Description**

Figure 10.8-1 depicts the CWS. The CWS is a recirculating system that supplies cooling water to the main condenser from the NHS. The circulating water pumps located in the intake structure of the NHS, take suction from the intake basin and discharge through a common underground

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

pipe. In the TB, the piping comes above grade and routes into the tube side of the main condenser to act as the cooling flow for condensing main turbine exhaust steam. Circulating water exiting the main condenser is piped back underground to the NHS.

A hot circulating water return line is also provided to recycle water returning from the condenser in cold weather conditions as required to prevent freezing in the NHS intake area and moderate cold water into the condenser to improve turbine performance. The CWS includes water box vents to help fill the condenser water boxes during startup and remove accumulated air and other gases from the water boxes during normal operation.

The PCW (Chapter 9, Subsection 9A.2.1) circulating water pumps are redundant and can provide water to the PCW heat exchanger at all times. During normal operation only one PCW heat exchanger is in-service, and the corresponding plant cooling water supply pump is used to provide cooling. During a plant shutdown, both PCW heat exchangers and PCW supply pumps can be in-service to facilitate a plant shutdown. With the loss of a PCW supply pump the plant shutdown may take longer due to the reduction in heat removal. In the event of a LOOP, the PCW portion of CWS is powered by the standby diesel generators to provide continuous cooling support to PCW.

The CWS drains the condenser water boxes via the water box drain pump. The CWS is designed for continuous operation and to provide sufficient circulating water to the condenser to support full power operation.

The CWS must be in operation before any steam or high energy sources are routed into the condenser. The CWS supplies cooling water at a flow rate that allows the steam entering the condenser the ability to condense.

A CWS pump may be taken out of service if the remaining pump is capable of supplying sufficient circulating water to cool the main condenser when operating at reduced load or the circulating water temperature is low enough.

The CWS, together with the NHS, provide circulating water to the main condenser in support of rated turbine generator power or greater operation. The CWS and NHS also supply circulating water to the main condenser sufficient to support 100% reactor power operation when operating up to 2% exceedance temperatures of the normal heat sink.

The PCW supply pump(s) must be in operation before any cooling water is required by the PCW. Only one pump is required to be in operation at a time to provide sufficient cooling capacity.

The PCW supply pumps provide sufficient cooling water to meet the performance requirements of the heat exchangers within the PCW. The PCW supply pumps are 100% capacity pumps. One pump can remain in standby while the operating pump provides the required cooling water flow.

Measure(s) are applied to control macrofouling (e.g., from zebra mussels) of the circulating cooling water systems.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSC.

#### **10.8.3.1 Component Description**

The following paragraphs describe major equipment items that comprise the CWS.

##### **10.8.3.1.1 Circulating Water Pumps**

The circulating water pumps are 50% capacity pumps. The pumps are arranged in parallel with suction from the NHS intake basin. There are two individual pump discharge lines and they combine into one circulating water supply line to the main condenser.

#### **10.8.3.1.2 Plant Cooling Water Supply Pumps**

The PCW supply pumps are 100% capacity pumps. These pumps take suction from the NHS intake basin.

The individual PCW supply pump discharge lines supply the independent PCW heat exchangers.

#### **10.8.3.1.3 Water Box Drain Pump**

The water box drain pump is sized to drain the CWS piping, pumps, and condenser water boxes. The water box drain pump discharge is routed to the CWS piping downstream of the condenser discharge isolation valves.

#### **10.8.3.1.4 Piping and Valves**

The CWS piping of 600 mm (24 in) nominal diameter and smaller is designed to meet the requirements of ASME B31.1, Power Piping (Reference 10.8-2). The piping with a nominal diameter of 650 mm (26 in) and larger is designed in accordance with American Water Works Association (AWWA) standards.

Valves installed at the discharge of the circulating water pumps are the motor-operated butterfly type. In the event of a trip of a circulating water pump, the closing characteristics of these valves limit the water hammer pressure to limits that are acceptable by the piping system.

All major CWS valves are operable by remote controls located in the MCR.

Underground piping from the circulating water pumps to the main condenser is made of a corrosion resistant material.

Pipe coating, lining, and cathodic protection is provided as appropriate based on-site specific conditions.

#### **10.8.3.1.5 Condenser Water Boxes**

The main condenser water boxes are constructed of carbon steel. The exterior surfaces are coated as appropriate based upon site-specific conditions.

The water box vacuum breakers open automatically to prevent excessive pressure transients caused by multiple circulating water pump trips.

Cathodic protection may be provided to protect the water boxes based on-site specific conditions.

### **10.8.4 Materials**

Material and equipment selection for the CWS is based on the range of environmental conditions that exist at the plant location.

### **10.8.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.8-1 for CWS interfaces with other systems.

### **10.8.6 System and Equipment Operation**

The CWS is designed for continuous operating for the following plant operating modes:

1. Normal operation
2. Off-normal operation

#### **10.8.6.1 Normal Operational Concept**

With two 50% capacity circulating water pumps in operation one of the two operating pumps may be manually stopped only if stopping the pump does not cause a trip of the turbine because of

HP in the condenser and that the pressure loss in the condenser with only one pump in operation is enough for the correct operation of its cleaning system. To accomplish this, power may need to be reduced to within the capacity of a single circulating water pump prior to stopping one pump.

Only one of two 100% capacity PCW pumps are in normal operation at a time with the other in standby. This does not preclude two pump operation under normal operations.

#### **10.8.6.2 Off-Normal Operational Concept**

The CWS is designed to maintain reactor operation during a turbine trip or a 25% load rejection, whereby 25% of the MS bypasses the turbine to the condenser.

##### ***10.8.6.2.1 Cold Weather Operation***

When required for cold weather operation, the system is designed to support cold weather startup by maintaining a minimum basin temperature. A hot-water recirculation line connects the piping downstream of the condenser with the intake forebay for use in maintaining the required intake temperature range.

Freeze protection is provided for above ground components that contain stagnant water.

##### ***10.8.6.2.2 Circulating Water Pump Trip***

If a circulating water pump trips, the associated pump discharge valve automatically closes to avoid water backflow. The trip of a circulating water pump alarms in the MCR. Tripping of the pump initiates a drop in power output and an increase in condenser back pressure due to an increase in circulating water temperature.

##### ***10.8.6.2.3 Plant Cooling Water Supply Pump Trip***

If a PCW supply pump trips, the backup pump auto starts and the process cooling water system will realign and put the associated heat exchanger in service. The trip of the pump alarms in the MCR.

##### ***10.8.6.2.4 Isolation of the Condenser Due to Tube Leak***

If a main condenser tube leak is detected, by changes in hotwell conductivity, the power output of the plant may need to be reduced to a value that accounts for the reduced cooling flow through the condenser tubes.

##### ***10.8.6.2.5 Flooding in the Turbine Building***

In the event that two of the three level transmitters in the condenser area detect a high-water level, a trip signal is sent to the circulating water pumps thereby tripping the pumps and the unit.

##### ***10.8.6.2.6 Isolation of the PCW System Due to High Radiation Levels***

In the event of high radiation levels detected downstream of either of the PCW heat exchangers, a signal is sent to the valves downstream of the heat exchangers. The valves close to prevent contaminated water from reaching the NHS.

#### **10.8.7 Instrumentation and Control**

The CWS provides the instrumentation to control and monitor (indicate and alarm) system operation.

##### **10.8.7.1 Instrumentation**

Major system parameters, i.e., forebay and water box levels; and process flow rate, pressures, and temperatures are indicated and alarmed as required to provide operational information and performance assessment.

#### **10.8.7.2 Control**

The CWS is operated and controlled from the MCR. Major system control functions, i.e., pump start/stop and valve positioning logics, are based on system operating parameters. For example, high radiation levels downstream of each PCW heat exchanger result in valve closure.

#### **10.8.8 Monitoring, Inspection, Testing, and Maintenance**

Routine testing of the CWS is conducted in accordance with normal power plant requirements for assuring component operability and integrity.

Equipment and components of the CWS are designed for ease of inspection and maintenance during plant operation. The CWS is provided with a water box drain pump with capability to drain the main condenser water boxes. Condenser inlet and outlet isolation block valves are provided to allow for isolation of condenser tube bundles for maintenance.

Each circulating water pump discharge has the ability to be individually isolated for service and maintenance. Manholes are provided for maintenance and cleaning of the circulating water piping and condenser water boxes.

#### **10.8.9 Radiological Aspects**

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.8.10 Performance and Safety Evaluation**

The CWS has two subsystems, the main condenser supply and the PCW supply. The CWS is capable of performing its Safety Category normal and off-normal functions as presented in Subsection 10.8.1. The CWS is capable of meeting its Safety Category 3 requirements.

The CWS provides sufficient circulating water to the main condenser to support full power operation. A pump may be taken out of service if the remaining pump is capable of supplying sufficient circulating water to cool the main condenser when operating at reduced load.

The PCW cooling water supply pumps are redundant and provide water to the PCW heat exchangers to remove PCW heat loads. The PCW cooling water supply pumps are powered by the standby diesel generators in the event of a LOOP to provide continuous cooling support to PCW. The PCW heat exchanger discharge is monitored to ensure the water returning to the NHS is free of radionuclides. High radiation levels would result in isolation of the discharge by automatic valve closure.

#### **10.8.11 References**

- 10.8-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 10.8-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.8-1: Circulating Water System Interfaces**

Interfacing System	Interface Description	Interface Boundary
SC2 and SC3 Instrumentation and Controls	SC2 and SC3 I&Cs provides control for the PCW supply pumps and associated valves	At CWS equipment
Non-Safety Instrumentation and Controls	Non-Safety I&Cs provides all Non-Safety I&C Control for pump, valves, and instrumentation	At CWS equipment
Radiation and Environmental Monitoring	Radiation monitoring of the PCW supply pump discharges	Downstream of the PCW heat exchangers
	Process sampling of the CWS pumps for chemistry control	Downstream of the main condenser tube bundles
Liquid Waste Management System	LWM outfall connects to CWS at the exit of the main condenser	At Condenser Water Box connection
Main Condenser and Auxiliaries	The CWS provides cooling water to the main condenser	At MCA Inlet/Outlet connections
Plant Cooling Water System	The CWS provides cooling water to PCW heat exchangers	At PCW heat exchanger Inlet/Outlet connections
Plant Pneumatics System	Provides instrument quality control air for control valves	At CWS AOVs
SC2 and SC3 Electrical Distribution System	Provides SC2 and SC3 related low and medium voltage power for the PCW supply pumps	At CWS equipment
Normal Heat Sink	The NHS provides lake water for the circulating water pumps	At NHS forebay and discharge structures



NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

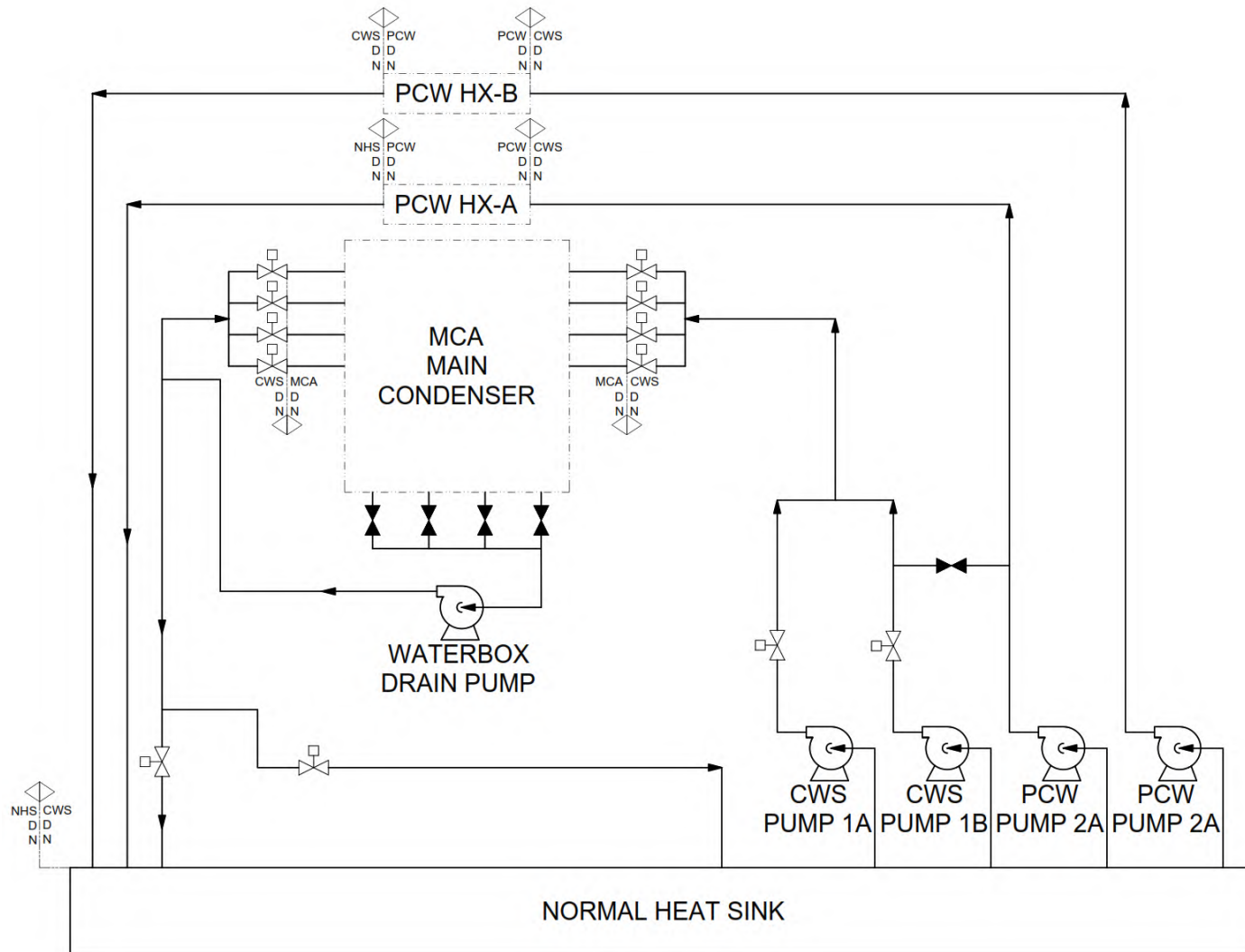


Figure 10.8-1: Circulating Water System

## **10.9 Generator and Exciter**

The Generator and Exciter System converts the rotational energy of the BWRX-300 turbine into electrical power that is transmitted to the Non-Safety Electrical Distribution System and utility power grid. The generator consists of a synchronous electric generator (including rotor, stator, and stator frame), collector, generator shaft seals, generator bearings, generator heat exchangers, neutral grounding resistor and associated neutral grounding transformer, potential transformers, current transformers, surge protection, generator protection panel and relaying (shared with Non-Safety Electrical Distribution System), the generator field winding, and instrumentation. Additionally, this system includes the Generator Excitation System that consists of a field exciter and Automatic Voltage Regulator that function to provide rated output voltage.

### **10.9.1 System and Equipment Functions**

#### **10.9.1.1 Normal Functions (Non-Safety Category)**

The function of the generator is to convert rotational energy produced by the main turbine into electrical power.

The function of the generator neutral grounding equipment is to provide protection for the generator in the event of a ground fault.

The Generator Excitation System does not perform any Non-Safety Category functions during normal conditions.

#### **10.9.1.2 Normal Functions (Safety Category)**

The functions of the generator exciter, in conjunction with the voltage regulator, are:

1. To supply direct current to the generator field winding to control the voltage and reactive volt-ampere output of the generator during all modes of power generation.
2. To operate in conjunction with the TGCS for synchronization during startup.
3. To operate in conjunction with the TGCS for power factor control during normal operation.
4. To operate in conjunction with the TGCS during shutdown.

The generator and exciter can initiate transients up to a scram.

#### **10.9.1.3 Off-Normal Functions (Non-Safety Category)**

This system does not perform any Non-Safety Category functions during off-normal conditions.

#### **10.9.1.4 Off-Normal Functions (Safety Category)**

This system does not perform any Safety Category functions during off-normal conditions.

### **10.9.2 Safety Design Bases**

The system has SC3 SSC and the neutral grounding transformer and resistor are SCN. The system is categorized as Non-Nuclear Seismic.

The generator provides input signals of system operating parameters to DCIS for monitoring.

The generator is able to withstand a sudden three-phase short circuit at the output terminals of the generator under a worst-case condition of the generator operating at full rated load, at rated power factor overexcited and up to the fabrication ratings of the machine.

Instrumentation is provided to monitor the operating parameters of the generator and excitation system over its anticipated ranges of normal and abnormal performance to facilitate plant operation and ensure safety.

The design of the MTE System meets CNSC requirements specified in REGDOC-2.5.2 Section 8.3.3 as it relates to providing provisions for overspeed protection, minimization of missiles due to a turbine break-up and inspection and testing (Reference 10.9-1). Minimization of potential missile impacts is also afforded through the favorable perpendicular orientation of the reactor to the turbine.

### **10.9.3 Description**

The generator produces its rated electrical output power at rated torque produced by the turbine and design condenser vacuum conditions.

The Generator and Exciter System provides electricity to be distributed to the grid via the isophase bus ducts and the generator step up transformer in Mode 1 and to the station loads via the Unit Auxiliary Transformer.

The generator operates at a 300 MW rating at 21 kV not more than five percent above or below rated voltage with a rated speed of 3600 rpm. The generator, exciter and generator control system are designed to be capable of responding to grid frequency changes.

The generator stator frame is fabricated of steel and is designed such that no damaging frame vibration occurs during normal operation and transients. The Generator Excitation System static exciter supplies and controls the direct current for the field winding of the generator and controls the voltage and reactive volt-ampere output of the generator. The DC current is supplied by a 250Vdc uninterruptible control power source. The neutral grounding is composed of a neutral grounding transformer with a secondary resistor and a current transformer dedicated to the stator earth fault protection. The Automatic Voltage Regulator monitors and adjusts the field excitation to maintain the output voltage within the specified range.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSC.

#### **10.9.3.1 Component Description**

##### **10.9.3.1.1 Generator**

The generator is capable of operating without air cooler cooling water flow at a level and duration determined by the manufacturer to not incur any damage. All the moving parts are provided with machine guards and barriers.

##### **10.9.3.1.2 Generator Exciter**

The generator exciter is designed to supply and control the direct current for the field winding of the generator and control the voltage and reactive volt-ampere output of the generator during normal plant operation and startup.

The power potential excitation transformer is protected by at least (and against):

- Transformer differential current
- Phase overcurrent
- Lightning strikes with appropriate surge protection and basic insulation level rating

##### **10.9.3.1.3 Automatic Voltage Regulator**

The voltage regulator, in conjunction with the generator exciter, maintains the output voltage of the generator at 21 kV +/- 5%.

#### **10.9.3.1.4 Neutral Ground Resistor**

The neutral grounding resistor is sized such that, in the event of a ground fault, the maximum current experienced by the neutral phase is between 3 and 25 amps.

#### **10.9.3.1.5 Neutral Ground Transformer**

The neutral grounding transformer in conjunction with the neutral grounding resistor, limits the maximum stator ground fault to between 3 and 25 amps.

#### **10.9.4 Materials**

Material and equipment selection for the Generator and Exciter System is based on the range of environmental conditions that exist at the plant location.

#### **10.9.5 Interfaces with Other Equipment or Systems**

Refer to Table 10.9-1 for information pertaining to Generator and Exciter System interfaces.

#### **10.9.6 System and Equipment Operation**

The generator is designed to operate during modes of normal power plant operation, including startup and shutdown activities. The generator is also designed to trip based on abnormal conditions. For all operating modes, automatic controls are provided for equipment and components to minimize operator action while ensuring optimum system performance. The system also operates in conjunction with the turbine auxiliary controller and generator auxiliary controller to perform system operations as needed.

##### **10.9.6.1 Startup**

The generator is controlled by the TGCS. The TGCS synchronizes the initial loading by:

- Matching turbine speed to the grid frequency
- Matching generator voltage to the grid
- Ensuring generator phase angle leads the grid phase angle
- Closing generator output breakers

##### **10.9.6.2 Normal Operation**

During normal operation, the TGCS controls the generator by operating in coordination with the Plant Automation System and RPC. During normal operation the TGCS utilizes the capabilities of the Automatic Voltage Regulator and the exciter to provide consistent generator output to the SCN Electrical Distribution System. The turbine and generator auxiliary controllers can also assist with the control of generator output as required.

##### **10.9.6.3 Off-Normal Operation**

During off-normal operation, the TGCS causes the generator to experience a load runback or trip of the generator, depending on the condition and the signals received from the Plant Automation System. These signals perform the following, based on conditions:

- Initiate runback or trip signals
- Provide signals for generator and switchyard breaker control, including tripping
- Runback the turbine generator to a safe operating condition based on input signals
- Provide automatic and manual controls to unload the unit during controlled shutdowns

## **10.9.7 Instrumentation and Control**

### **10.9.7.1 Instrumentation**

Instrumentation is provided to monitor the operating parameters of the generator and excitation system over anticipated ranges of normal and abnormal performance to facilitate plant operation and ensure safety.

Manual and automatic controls are provided to maintain generator parameters within prescribed operating ranges and, specifically, to control the voltage and reactive volt-ampere output of the generator.

The following alarms and trip signals are generated based on inputs provided to the DCIS:

- A. Stator winding fault
- B. Loss of excitation
- C. Negative phase sequence currents
- D. Stator overcurrent
- E. Stator overvoltage/undervoltage
- F. Loss of synchronism
- G. Journal bearing vibration high
- H. Reverse power (time delayed)
- I. Field overexcitation
- J. Under-frequency
- K. Neutral overvoltage
- L. Distance Backup Protection
- M. Inadvertent Energization
- N. Generator Vibration High
- O. Sequential Trip signals (Dry Contacts and Power Contacts)

Upon sensing any of the following abnormal generator conditions, an alarm is actuated in the MCR based on generator parameters that are provided to the DCIS:

- A. Rotor winding ground
- B. Stator winding temperatures
- C. Stator winding cooling coil outlet temperatures
- D. Journal bearing metal temperatures
- E. Shaft voltage
- F. Collector-ring cooling-air inlet and outlet temperatures
- G. Generator bushing temperatures
- H. Core-end or flux-shield temperatures
- I. Air cooler gas outlet temperature
- J. Core monitor conditions

- K. Generator voltage
- L. Generator frequency
- M. Generator output volts/hertz
- N. Excitation field ground
- O. DC Excitation potential

A zero (0) speed indication signal is supplied to reset the alarm for generator protection equipment or generator health monitoring equipment.

The following signals are sent from the static exciter to the generator protection panel:

- A. Generator voltage
- B. Generator DC excitation voltage
- C. Generator DC excitation current
- D. Volts/Hertz
- E. Power potential transformer lockout relay status
- F. Instantaneous DC current
- G. Bridge temperature
- H. Bridge AC phase angle
- I. Phase voltage

The power potential excitation transformer is protected by at least (and against) the following:

- A. Transformer differential current
- B. Phase overcurrent
- C. Lightning strikes with appropriate surge protection and basic insulation level rating

#### **10.9.7.2 Control**

As part of the Generator and Exciter System design, controls and protection signals are capable of being multiplexed based on interfacing system requirements. Fire Protection signals are provided to the Main Fire Alarm Panel and the DCIS and can be used in the coordination with other plant equipment and all cyber security requirements are followed.

The generator protection system design includes redundant instrumentation with a coincident logic trip system. Initiation devices for all trips have a separate set of contacts that can be used for DCIS monitoring. DCIS Human Machine Interface screen displays are provided for each trip initiating signal to identify the source of the trip signal.

#### **10.9.8 Monitoring, Inspection, Testing, and Maintenance**

The equipment and components of the generator are designed for ease of inspection and maintenance during plant operation. Generator and Exciter System components are arranged, sized, and selected to facilitate their replacement during a normal maintenance outage. Individual components are manufactured to standard sizes and gauges to the greatest extent possible.

Routine testing of the generator is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. An inspection of the generator is performed per the vendor's supplied maintenance and inspection requirements.

The Generator and Exciter System supports online inspection activities to the extent practical. Condition monitoring equipment and instrumentation is included where appropriate to ensure equipment performance and health is as desired. Maintenance can be performed online where redundant components can be bypassed or taken out of service one at a time without affecting plant operation.

#### **10.9.9 Radiological Aspects**

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states.

#### **10.9.10 Performance and Safety Evaluation**

The Generator and Exciter System is provided with sufficient I&C channels such that the system can continue operation in the event of the failure of a single instrumentation channel. The generator protection system design includes redundant instrumentation with a coincident logic trip system.

#### **10.9.11 References**

- 10.9-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

NEDO-33960 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 10.9-1: Generator and Exciter System Interfaces**

Interfacing System	Interface Description	Interface Boundary
SC2 and SC3 Instrumentation and Control System	Monitoring of Generator and Exciter System bearing vibration. Performs steam turbine and generator protection functions	Turbine Generator Control Equipment
Non-Safety Instrumentation and Control System	Monitoring of Generator and Exciter System parameters returned for indication purposes	Turbine Generator Control Equipment
Main Turbine Equipment	Generator Rotor connected to turbine, pass forces back and forth along the turbine/generator drivetrain. Includes system connections	Generator Rotor and Main Turbine Shaft coupling; Lube oil supply and discharge connections
Plant Cooling Water System	Provides cooling to generator air coolers	Inlet and outlet connections
SC2 and SC3 Electrical Distribution System	Provides power to Automatic Voltage Regulator and Excitation Cubicle	Automatic Voltage Regulator and Excitation Cubicle
Non-Safety Electrical Distribution System	Provides power to the Unit Auxiliary Transformer and Generator step up for distribution to the plant and switchyard, respectively	Connection between Generator Terminals and Isophase Equipment Terminals Generator Protective Equipment and Relaying
Heating Ventilation and Cooling System	Generator and Exciter System is air cooled by building heat removal	Exterior shell of generator and excitation cabinets
Turbine Building Structure	Equipment is supported by the building structure	TB Structure



#### **10.10 Implementation of Break Preclusion for the Main Steam and Feedwater Lines**

Refer to Chapter 3, Subsection 3.4.4 for information pertaining to break exclusion.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33961

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 11  
Management of Radioactive Waste**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARM	Area Radiation Monitoring Subsystem
BWR	Boiling Water Reactor
CFD	Condensate Filters and Demineralizers System
CFS	Condensate and Feedwater Heating System
CIV	Containment Isolation Valve
CMon	Containment Monitoring
CO	Carbon Monoxide
CRD	Control Rod Drive
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
EFS	Equipment and Floor Drain System
FPC	Fuel Pool Cooling and Cleanup System
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container
HVAC	Heating, Ventilation and Air Conditioning
HVS	Heating, Ventilation and Cooling System
IC	Isolation Condenser
ICC	ICS Pool Cooling and Cleanup System
ICS	Isolation Condenser System
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
MCR	Main Control Room
MSL	Main Steam Line
NBS	Nuclear Boiler System
NS	Non-Safety
OGS	Offgas System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
RB	Reactor Building
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SC	Safety Class
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejectors SJAE
SWM	Solid Waste Management System
TB	Turbine Building
WGC	Water, Gas and Chemical Pads

## TABLE OF CONTENTS

<b>11.0 MANAGEMENT OF RADIOACTIVE WASTE.....</b>	<b>11-1</b>
11.1 Source Terms.....	11-2
11.1.1 Coolant Radiation Concentrations .....	11-2
11.1.2 Fission Products .....	11-3
11.1.3 Activation Products .....	11-4
11.1.4 Radionuclide Concentration Adjustments .....	11-5
11.1.5 References .....	11-7
11.2 Systems for Management of Liquid Radioactive Waste .....	11-18
11.2.1 System and Equipment Functions .....	11-18
11.2.2 Safety Design Bases .....	11-18
11.2.3 Description .....	11-19
11.2.4 Materials.....	11-20
11.2.5 Interfaces with Other Equipment or Systems .....	11-20
11.2.6 System and Equipment Operation .....	11-20
11.2.7 Instrumentation and Control.....	11-21
11.2.8 Monitoring, Inspection, Testing, and Maintenance .....	11-22
11.2.9 Radiological Aspects .....	11-22
11.2.10 Performance and Safety Evaluation.....	11-23
11.2.11 References .....	11-23
11.3 System for Management of Gaseous Radioactive Waste .....	11-27
11.3.1 System and Equipment Functions .....	11-27
11.3.2 Normal Functions (Non-Safety Category) .....	11-27
11.3.3 Normal Functions (Safety Category) .....	11-27
11.3.4 Off-Normal Functions (Non-Safety Category) .....	11-27
11.3.5 Off-Normal Functions (Safety Category) .....	11-27
11.3.6 Safety Design Bases .....	11-27
11.3.7 Description .....	11-27
11.3.8 Materials.....	11-31
11.3.9 Interfaces with Other Equipment or Systems .....	11-31
11.3.10 System and Equipment Operation .....	11-31
11.3.11 Instrumentation and Control.....	11-32
11.3.12 Monitoring, Inspection, Testing, and Maintenance .....	11-33
11.3.13 Radiological Aspects .....	11-34

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

11.3.14	Performance and Safety Evaluation.....	11-34
11.3.15	References .....	11-34
11.4	System for Management of Solid Radioactive Waste .....	11-41
11.4.1	System and Equipment Functions .....	11-41
11.4.2	Safety Design Bases .....	11-41
11.4.3	Description .....	11-41
11.4.4	Materials.....	11-43
11.4.5	Interfaces with Other Equipment or Systems .....	11-43
11.4.6	System and Equipment Operation .....	11-43
11.4.7	Instrumentation and Control.....	11-44
11.4.8	Monitoring, Inspection, Testing, and Maintenance .....	11-44
11.4.9	Radiological Aspects .....	11-45
11.4.10	Performance and Safety Evaluation.....	11-45
11.4.11	References .....	11-45
11.5	Process and Effluent Radiological Monitoring.....	11-48
11.5.1	System and Equipment Functions .....	11-48
11.5.2	Safety Design Bases .....	11-49
11.5.3	Description .....	11-50
11.5.4	Materials.....	11-50
11.5.5	Interfaces with Other Equipment or Systems .....	11-50
11.5.6	System and Equipment Operation .....	11-53
11.5.7	Instrumentation and Control.....	11-53
11.5.8	Monitoring, Inspection, Testing, and Maintenance .....	11-53
11.5.9	Radiological Aspects .....	11-53
11.5.10	Performance and Safety Evaluation.....	11-54
11.5.11	References .....	11-54



## LIST OF TABLES

Table 11.1-1: Source Term Design Basis Parameters.....	11-8
Table 11.1-2a: Design Basis Noble Radiogas Source Terms in Steam .....	11-9
Table 11.1-2b: Normal Operational Noble Radiogas Source Terms in Steam .....	11-10
Table 11.1-3: Calculational Parameters for Source Term Adjustment .....	11-11
Table 11.1-4a: Design Basis Iodine Radioisotopes in Reactor Water and Steam.....	11-12
Table 11.1-4b: Normal Operational Iodine Radioisotopes in Reactor Water and Steam.....	11-13
Table 11.1-5a: Design Basis Non-Volatile Fission Products in Reactor Water and Steam ..	11-14
Table 11.1-5b: Normal Operational Non-Volatile Fission Products in Reactor Water and Steam .....	11-15
Table 11.1-6: N16 Concentrations in Reactor Water and Steam*** .....	11-16
Table 11.1-7: Activated Corrosion Products in Reactor Water and Steam* .....	11-17
Table 11.2-1: System Interfaces .....	11-24
Table 11.3-1: System Interfaces .....	11-35
Table 11.3-2: BWRX-300 Operating Modes .....	11-38
Table 11.3-3: BWRX-300 Offgas System Failure Dose Results .....	11-39
Table 11.4-1: SWM System Interfaces .....	11-46
Table 11.5-1: PRM Subsystem System Interfaces .....	11-55

**LIST OF FIGURES**

Figure 11.2-1: LWM Simplified Flow Diagram .....	11-25
Figure 11.2-2: Condensate Storage Tank Users Simplified Flow Diagram .....	11-26
Figure 11.3-1: OGS Simplified Flow Diagram.....	11-40
Figure 11.4-1: System Simplified Line Diagram.....	11-47

## **11.0 MANAGEMENT OF RADIOACTIVE WASTE**

This chapter describes the main sources of liquid, gaseous and solid radioactive waste including the radiological source term used in calculating liquid and airborne effluent. Also described are the radioactive waste processing systems (i.e., pre-treatment, treatment, and conditioning systems) as well as temporary waste storage located on the site. The design of the Structures, Systems and Components (SSC) provided for these purposes and the instrumentation provided to monitor for possible leaks of radioactive waste is described per Canadian Nuclear Safety Commission (CNSC) REGDOC-2.5.2, Section 8.11 (Reference 11.1-1).

The systems and equipment that monitor and sample the process and effluent streams to measure and control the discharge of radioactive materials generated in operational states and accident conditions are described.

The measures for the safe management of radioactive and hazardous waste of all types that will be generated throughout the lifetime of the plant as well as how these measures meet the relevant safety requirements including the measures taken for the safe management and disposal of this waste are described in OPG's Waste Management Program.

### **Sources of Waste**

Liquid radioactive waste is generated by cleanup of the reactor coolant, cleanup of the Spent Fuel Pool (SFP), and leakage from process streams. Leaked liquids are collected through floor drains, pump bed plate drains, and process drains in the Reactor Building (RB), Turbine Building (TB), Radwaste Building (RWB) and Plant Services Area. These waste sources are collected by drains and drain headers, routed by gravity drainage piping to a sump, and pumped to the Liquid Waste Management System (LWM) for processing.

Gaseous radioactive waste is generated as a result of activation of the reactor coolant during normal reactor operation, tramp uranium that may adhere to the outside of the fuel cladding during fabrication, and leakage of fission gas through miniscule defects in the fuel cladding.

Solid waste is created through plant operations, decontamination activities and through processing liquid radwaste. It includes the filter backwash sludges, reverse osmosis concentrates, charcoal media, and bead resins generated by the Fuel Pool Cooling and Cleanup (FPC), Isolation Condenser System Pool Cooling and Cleanup System (ICC), and Condensate Filters and Demineralizer (CFD) systems. Contaminated solids such as High Efficiency Particulate Air (HEPA) and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also disposed of in the Solid Waste Management System (SWM).

### **Waste Minimization**

The BWRX-300 is designed to minimize the generation of radioactive waste to the extent practicable as per CNSC REGDOC-2.5.2, Section 8.13.4 (Reference 11.1-1).

The BWRX-300 LWM is designed such that under normal conditions the need to release liquid effluent to the environment is minimized. LWM tanks are designed with significant holdup capacity and filtration skids are capable of filtering water to a sufficient quality for return to plant processes.

The Offgas System (OGS), described in Section 11.3, minimizes, and controls the release of radioactive material into the atmosphere by delaying and filtering the offgas process steam containing the radioactive isotopes so that gaseous effluent levels are below action levels calculated per CSA Group (CSA) N288.8 (Reference 11.1-2) ensuring prescribed limits for effluent releases cited in SOR/2000-203 (Reference 11.1-3) are not exceeded before discharge from the plant.

As discussed in Section 11.5, the Process and Effluent Monitoring System minimizes the generation of radioactive waste by directing continuous samples from radioactive processes back to the sampled process and by minimizing the amount of a sample that needs to be extracted, consistent with laboratory and sensitivity requirements.

### **Waste Storage**

Gaseous and liquid waste generated through normal plant processes are held up and filtered to a sufficient quality to allow for return to plant processes or released as described above. The design is such that there is no need for storage of gaseous or liquid waste external to the systems.

To the extent possible, discarded chemicals (including those classified as hazardous) are kept separate from solid radioactive waste. Section 11.4 discusses how solid waste is categorized, separated, and prepared for short term storage and eventual transport to an offsite facility.

### **11.1 Source Terms**

Two source terms are presented in this section. The first source term is a realistic model used to predict expected long-term average concentrations of radionuclides in the primary and secondary fluid streams. The second is a conservative design basis model that utilizes an established GEH fuel clad defect model, Reference 11.1-5.

The realistic model, which is based on available measured nuclide concentrations during normal operation at operating BWRs, is formulated as a standard for the American Nuclear Standard Source Term Specifications (ANSI) and American Nuclear Society (ANS) 18.1 (Reference 11.1-4) and is the source term model used to estimate liquid and gaseous effluent releases. The realistic model is referred to as the normal operation source term.

The second source term conservatively employs a GEH fuel clad defect model, Reference 11.1-5, serves as a basis for system and shielding requirements and is referred to as the design basis coolant source term.

#### **11.1.1 Coolant Radiation Concentrations**

##### **Normal Operation Coolant Source Term**

The normal operation BWRX-300 reactor water and steam radiation concentrations are based on Boiling Water Reactor (BWR) operating data reported in the ANSI/ANS-18.1 (Reference 11.1-4). This standard provides bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR. The source term data is based on the cumulative industry experience at operating BWR plants, and measurements at several stations. It therefore reflects the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs. The source terms specified in this section were obtained by applying the procedures of Reference 11.1-4 for estimation of typical source terms and adjusting the results, as appropriate, to assure conservative bases for design.

The BWRX-300 cleanup system configuration differs from the reference configuration in ANSI/ANS-18.1, and typical BWRs, since the CUW does not include any filtering media. The CFD performs all of the cleanup of the reactor coolant and is expected to filter the reactor coolant more efficiently than existing BWR systems employing a traditional reactor water cleanup system with a condensate demineralizer. The condensate filters include three filter vessels with high efficiency backwash-type filter elements arranged in parallel, and the condensate demineralizers include three demineralizer vessels arranged in parallel which significantly increases the residence time of the of the condensate and represents the best available technology economically achievable for the CFD.

A performance assumption is established at this stage of the design process that assumes the BWRX-300 CFD will provide coolant cleanup that is equivalent or better than existing BWRs that employ the use of traditional coolant cleanup systems. Thus, the ANSI/ANS-18.1 cleanup system removal efficiencies based on BWR operating data are used for the source terms presented here and must be validated when a CFD vendor is selected.

The information provided in this section defines the radioactive source terms in the reactor water and steam which serve as design bases for the gaseous, liquid, and solid radioactive waste management systems.

### **Design Basis Coolant Source Term**

The radionuclides included in the design basis source term are categorized as fission products or activation products. The lists do not necessarily include all radionuclides that may be detectable or theoretically predicted to be present. The principal nuclides identified in ANSI/ANS 18.1 are included because they are potentially significant with respect to one or more of the following criteria:

- Plant equipment design
- Shielding design
- Understanding system operation and performance
- Measurement practicability

#### **11.1.2 Fission Products**

##### **Normal Operation Noble Radiogas**

The BWRX-300 normal operation noble gas source term consists of the thirteen principle noble gas fission products as observed in steam flowing from the reactor vessel which are provided in the Radioactive Source Term Standard ANSI/ANS-18.1 (Reference 11.1-4). Concentrations in the reactor water are considered negligible under normal power operation because all of the gases released to the coolant are assumed to be rapidly transported out of the vessel with the steam and removed from the system with the other non-condensable gases in the main condenser. Due to the rapid removal of the gases, the expected relative mix of gases does not depend on the reactor design. The normal operation principle noble radiogas source terms are presented in Table 11.1 2b.

##### **Design Basis Noble Radiogas**

The design basis noble gas source terms are developed by normalizing the ANSI/ANS-18.1 normal operation noble gas source term mixture to a fuel release rate of 3,700 MBq/sec (100,000  $\mu$ Ci/sec) at 30 minutes decay. The noble radiogas source term rate after 30-minute decay has been used as a conventional measure of the fuel leakage rate, since it is conveniently measurable and was consistent with the nominal 30-minute offgas holdup system used on a number of early plants. A design basis noble gas release rate of 3700 MBq/s at 30-minute decay has historically been used for the design of the gaseous waste treatment systems in BWR plants (Reference 11.1-5) with satisfactory results. It was selected on the basis of operating experience with consideration given to several judgmental factors, including the implications to environmental releases, system contamination, and building air contamination. The design basis principle noble radiogas source terms are presented in Table 11.1-2a.

##### **Normal Operation Radioiodine**

The BWRX-300 normal operation radioiodine concentrations of the 5 principal radioiodines in reactor water and steam are determined by adjusting BWR Reference Plant radioiodine

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

concentrations using the methods in ANSI/ANS-18.1 and the BWRX-300 operating parameters in Table 11.1-4b.

### **Design Basis Radioiodine**

Historically, design basis radioiodine source terms for BWRs are normalized to an I-131 leak rate of 26 MBq/sec (700  $\mu$ Ci/sec) from the fuel (Reference 11.1-5). Experience indicates that I-131 leakage rates this high would only be approached during full power operation with multiple fuel cladding defects. It is not expected that full power operation would continue for any significant period of time with fuel cladding defects as severe as indicated by I-131 leakage in excess of 26 MBq/sec (700  $\mu$ Ci/sec).

The design basis reactor water radioiodine concentrations are developed by normalizing the ANSI/ANS-18.1 normal operation radioiodine concentrations to an I-131 leakage rate of 26 MBq/sec (700  $\mu$ Ci/sec) from the fuel (Reference 11.1-5) reported in Table 11.1-1. This provides a margin relative to the expected I-131 release rate shown in Table 11.1-1.

The design basis concentrations are presented in Table 11.1-4a.

### **Other Fission Products**

This category includes fission products other than noble gases and iodines and also includes transuranic nuclides. Some of the fission products are noble gas daughter products that are produced in the steam and condensate system. The only transuranic which is detectable in significant concentrations is Np-239.

The BWRX-300 normal operation concentrations for fission products other than noble gases and iodine in reactor water and steam are determined by adjusting BWR Reference Plant radioiodine concentrations using the methods in ANSI/ANS-18.1 and the BWRX-300 operating parameters in Table 11.1-3. In order to assure conservative design basis concentrations for the BWRX-300, the normal operation is normalized to the I-131 leakage rate from the fuel reported in Table 11.1-1.

The design basis reactor water concentrations are presented in Table 11.1-5a, and the normal operation concentrations reside in Table 11.1-5b.

### **11.1.3 Activation Products**

#### **11.1.3.1 Coolant Activation Products**

The coolant activation product of primary importance in BWRs is N-16. Based on the ANSI/ANS 18.1 (Reference 11.1-4) BWR Reference Plant and the BWRX-300 operating parameters, the N-16 levels of in steam leaving the reactor vessel for the plant without Hydrogen Water Chemistry system (HWC) is 1.78 MBq/gm (48  $\mu$ Ci/gm), and 9.25 MBq/gm (250  $\mu$ Ci/gm) with HWC. The BWRX-300 is expected to utilize HWC. This is treated as essentially independent of reactor design because both the production rate of N-16 and the steam flow rate from the vessel vary in direct proportion to reactor thermal power. For this reason, the same N-16 concentrations in reactor water and steam with HWC is used for both the normal operation and the design basis source terms.

It should be noted that a portion of the source term traditionally identified as "N-16" actually represents C-15. To the extent that C-15 is present, it is generally about ~0.55 MBq/gm (15  $\mu$ Ci/gm) or less. Historically, gross gamma dose rate measurements made to confirm the magnitude of the N-16 concentration have included responses to gamma rays from C-15. Use of the combined "N-16" source term in shielding design introduces additional conservatism because the C-15 component has a 2.45 second half-life, and therefore decays more rapidly with transport time through the system than N-16, which has a 7.1 second half-life.

The design basis N-16 concentrations in steam and reactor water are shown in Table 11.1-6. Reference 11.1-4 gives the reactor water concentration at the recirculation system. Because the BWRX-300 does not have an external recirculation loop, the reactor water concentration has been decay-corrected to the reactor core exit to obtain an estimated value shown in Table 11.1-1.

#### **11.1.3.2 Non-Coolant Activation Corrosion Products**

Radionuclides produced in the coolant by neutron activation of circulating impurities and by corrosion of irradiated system materials are classified as activated corrosion products. Typical reactor water concentrations for activated corrosion products are contained in the ANSI/ANS 18.1 BWR Reference Plant concentrations (Reference 11.1-4).

The BWRX-300 activated corrosion product concentrations are calculated by adjusting the BWR Reference Plant concentrations of non-coolant activation products to BWRX-300 conditions by using the methods in ANSI/ANS 18.1 and appropriate data from Table 11.1-3.

The BWRX-300 design basis concentrations of the activated corrosion products are not normalized using the fuel leakage model because they do not originate in the fuel but are rather produced by corrosion of irradiated system materials in the reactor and associated piping. Thus, the production of activated corrosion products is dependent on the inner surface area of the reactor and piping where gamma and neutron activation can occur, and the BWRX-300 has much less surface area than the large BWRs that provide the operating data bases for BWR Reference Plant concentrations.

The activated corrosion products concentrations in reactor water and steam are presented in Table 11.1-7. The Table 11.1-7 concentrations are used for both normal operation and design basis source terms.

#### **Tritium**

Tritium is produced by activation of naturally occurring deuterium in the primary coolant and, to a lesser extent, as a fission product in the fuel (Reference 11.1-6). The tritium is primarily present as tritiated oxide, T-O-H. Because tritium has a long half-life (12 years) and is not affected by cleanup processes in the system, the concentration is controlled by the rate of loss of water from the system by evaporation or leakage (Reference 11.1-4). Plant process water and steam have a common tritium concentration. The concentration reached depends on the actual water loss rate; however, Reference 11.1-4 specifies a typical concentration of 518 Bq/gm (0.014  $\mu$ Ci/gm) based on BWR operating data adjusted to account for liquid recycle. This value is applicable for the BWRX-300.

#### **Argon-41**

Argon-41 is produced in the reactor coolant as a consequence of neutron activation of naturally occurring Argon-40 in air that is entrained in the feedwater. The Argon-41 gas is carried out of the vessel with the steam and stripped from the system with the non-condensable gases in the main condenser. Observed Argon-41 levels are highly variable due to the variability in air in-leakage rates into the system. Reference 11.1-6 specifies a normal operation Argon-41 release rate from the vessel into the offgas treatment system of 1.5 MBq/sec (40  $\mu$ Ci/sec). This value is considered conservative as it bounds the available experimental database; this value is provided in Table 11.1-1..

#### **11.1.4 Radionuclide Concentration Adjustments**

In order to determine the estimated concentrations of radionuclides in the groups classified as iodines, other non-volatile fission products, and non-coolant activation products using the ANSI/ANS 18.1 Source Term Standard (Reference 11.1-4), it is necessary to apply appropriate

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

adjustment factors to the Reference Plant concentrations provided in the Standard, per the following equations.

Equilibrium concentrations in reactor water are assumed to satisfy the relationship:

$$C = \frac{S}{M(\lambda + R)}$$

where:

- C = radionuclide concentration
- S = radionuclide input rate to coolant
- M = reactor water mass
- $\lambda$  = radionuclide decay constant
- R = sum of removal rates of the radionuclide from the system.

Consequently, if the radionuclide input rate is taken to depend primarily on the reactor thermal power, the adjustment factors to be applied to the Reference Plant reactor water concentrations are given by:

(11.1-2)

$$\text{Adjustment Factor} = \frac{P M_r (\lambda + R_r)}{P_r M (\lambda + R)}$$

where the subscript "r" refers to the Reference Plant, P is the reactor thermal power and M,  $\lambda$ , and R are as defined above.

The removal rate from the system is the sum of the removal rates due to the CUW and the condensate demineralizer and is given by:

$$R = \frac{(FA \cdot NA) + (NC \cdot FS \cdot NS \cdot NB)}{WP} \quad (11.1-3)$$

where:

- FA = CUW flow rate
- NA = fraction of nuclide removed in CUW
- NC = ratio of CFD System flow rate to steam flow rate (=1 for BWRX-300)
- FS = steam flow rate
- NS = ratio of nuclide concentration in steam to concentration in water
- NB = fraction of nuclide removed by the CFD System
- WP = weight of water in the reactor vessel

The Reference Plant and BWRX-300 plant parameters and the nuclide-dependent removal rate parameters used for the BWRX-300 are shown in Table 11.1-3.



NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**11.1.5 References**

- 11.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 11.1-2 CSA N288.8, "Establishing and implementing action levels for releases to the environment from nuclear facilities," CSA Group.
- 11.1-3 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 11.1-4 ANSI/ANS-18.1, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society.
- 11.1-5 NEDO-10871, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms," GE-Hitachi Nuclear Energy Americas, LLC.
- 11.1-6 USNRC NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR)."

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-1: Source Term Design Basis Parameters**

Parameter	Value
Total of the design basis release rates of the 13 noble gases (30-minute decay reference, t=30)	3700 MBq/sec (100,000 $\mu$ Ci/sec)
Design basis I-131 radioiodine core release rate	26 MBq/sec (700 $\mu$ Ci/sec)
Reactor core exit N-16 concentration (design basis same as normal operation)	1.85 MBq/gm (50 $\mu$ Ci/gm) w/o HWC 9.25 MBq/gm (250 $\mu$ Ci/gm) w/HWC
Normal operational Ar-41 release rate into the offgas treatment system	1.5 MBq/sec (40 mCi/sec)

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-2a: Design Basis Noble Radiogas Source Terms in Steam**

Isotope	Decay Constant (per hour)	Steam Concentration		Source Term at t=30min	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/sec)	( $\mu$ Ci/sec)
Kr-83m	3.73E-1	2.4E-03	6.5E-02	1.0E+03	2.7E+04
Kr-85m	1.55E-1	6.6E-05	1.8E-03	3.1E+01	8.3E+02
Kr-85	7.37E-6	1.6E-05	4.4E-04	8.3E+00	2.2E+02
Kr-87	5.47E-1	4.5E-04	1.2E-02	1.7E+02	4.7E+03
Kr-88	2.48E-1	2.5E-04	6.9E-03	1.1E+02	3.1E+03
Kr-89	1.32E+1	8.6E-02	2.3E+00	6.0E+01	1.6E+03
Xe-131m	2.41E-3	1.4E-05	3.7E-04	6.8E+00	1.8E+02
Xe-133m	1.30E-2	8.2E-06	2.2E-04	4.1E+00	1.1E+02
Xe-133	5.46E-3	1.2E-04	3.3E-03	6.2E+01	1.7E+03
Xe-135m	2.72E+0	2.2E-03	6.0E-02	2.9E+02	7.7E+03
Xe-135	7.56E-2	1.2E-03	3.3E-02	6.0E+02	1.6E+04
Xe-137	1.08E+1	4.1E-03	1.1E-01	9.1E+00	2.5E+02
Xe-138	2.93E+0	1.1E-02	3.1E-01	1.3E+03	3.6E+04

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-2b: Normal Operational Noble Radiogas Source Terms in Steam**

Isotope	Decay Constant (per hour)	Steam Concentration	
		(MBq/gm)	( $\mu$ Ci/gm)
Kr-83m	3.73E-1	2.2E-05	5.9E-04
Kr-85m	1.55E-1	5.9E-07	1.6E-05
Kr-85	7.37E-6	1.5E-07	4.0E-06
Kr-87	5.47E-1	4.1E-06	1.1E-04
Kr-88	2.48E-1	2.3E-06	6.2E-05
Kr-89	1.32E+1	7.8E-04	2.1E-02
Xe-131m	2.41E-3	1.2E-07	3.3E-06
Xe-133m	1.30E-2	7.4E-08	2.0E-06
Xe-133	5.46E-3	1.1E-06	3.0E-05
Xe-135m	2.72E+0	2.0E-05	5.4E-04
Xe-135	7.56E-2	1.1E-05	3.0E-04
Xe-137	1.08E+1	3.7E-05	9.9E-04
Xe-138	2.93E+0	1.0E-04	2.8E-03

**Table 11.1-3: Calculational Parameters for Source Term Adjustment**

<b>A. Plant Parameters for Source Term Adjustment</b>		
<b>Parameter</b>	<b>Reference Plant</b>	<b>BWRX-300</b>
Thermal Power, MWt	3400	870
Reactor Water Mass, kg (lb)	1.70E+5 (3.80E+5)	1.23E+05 (2.72E+05)
CUW Flow Rate, kg/hr (lb/hr)	5.80E+4 (1.30E+5)	About 1% of the steam flow rate by design
Steam Flow Rate, kg/hr (lb/hr)	6.80E+6 (1.50E+7)	5.03E+02 (3.99E+06)
Ratio of Condensate Demineralizer Flow Rate to Steam Flow Rate	1	1

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-4a: Design Basis Iodine Radioisotopes in Reactor Water and Steam**

Isotope	Water Concentration		Steam Concentration	
	(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
I-131	1.9E-03	5.1E-02	3.9E-05	1.0E-03
I-132	2.0E-02	5.5E-01	3.9E-04	1.0E-02
I-133	1.4E-02	3.8E-01	2.8E-04	7.7E-03
I-134	6.2E-02	1.7E+00	1.2E-03	3.3E-02
I-135	2.8E-02	7.6E-01	5.4E-04	1.5E-02

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-4b: Normal Operational Iodine Radioisotopes in Reactor Water and Steam**

Isotope	Decay Constant (per hour)	Water Concentration		Steam Concentration	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
I-131	3.59E-03	2.7E-06	7.4E-05	5.6E-08	1.5E-06
I-132	3.03E-01	3.0E-05	8.0E-04	5.7E-07	1.5E-05
I-133	3.33E-02	2.1E-05	5.6E-04	4.1E-07	1.1E-05
I-134	7.91E-01	9.0E-05	2.4E-03	1.8E-06	4.9E-05
I-135	1.05E-01	4.1E-05	1.1E-03	7.9E-07	2.1E-05

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-5a: Design Basis Non-Volatile Fission Products in Reactor Water and Steam**

Isotope*	Decay Constant (per hour)	Water Concentration		Steam Concentration	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
Rb-89	2.74E+00	4.7E-02	1.3E+00	4.7E-05	1.3E-03
Sr-89	5.55E-04	6.3E-05	1.7E-03	6.3E-08	1.7E-06
Sr-90	2.81E-06	3.2E-06	8.5E-05	3.2E-09	8.5E-08
Y-90	2.81E-06	3.2E-06	8.5E-05	3.2E-09	8.5E-08
Sr-91	7.31E-02	4.1E-02	1.1E+00	4.1E-05	1.1E-03
Sr-92	2.56E-01	9.4E-02	2.5E+00	9.4E-05	2.5E-03
Y-91	4.93E-04	8.4E-04	2.3E-02	8.4E-07	2.3E-05
Y-92	1.96E-01	2.8E-02	7.7E-01	2.8E-05	7.7E-04
Y-93	6.80E-02	2.9E-03	7.9E-02	2.9E-06	7.9E-05
Zr-95/Nb-95	4.41E-04	1.7E-03	4.7E-02	1.7E-06	4.7E-05
Mo-99/Tc-99m	1.05E-02	8.3E-03	2.2E-01	8.3E-06	2.2E-04
Ru-103/Rh-103m	7.29E-04	4.2E-04	1.1E-02	4.2E-07	1.1E-05
Ru-106/Rh-106	7.83E-05	6.3E-05	1.7E-03	6.3E-08	1.7E-06
Te -129m	8.65E-04	8.4E-04	2.3E-02	8.4E-07	2.3E-05
Te-131m	2.31E-02	1.1E-03	3.0E-02	1.1E-06	3.0E-05
Te-132	8.89E-03	2.0E-04	5.5E-03	2.0E-07	5.5E-06
Cs-134	3.84E-05	5.7E-04	1.5E-02	5.7E-07	1.5E-05
Cs-136	2.22E-03	4.5E-04	1.2E-02	4.5E-07	1.2E-05
Cs-137/Ba-137m	2.63E-06	8.7E-04	2.4E-02	8.7E-07	2.4E-05
Cs-138	1.29E+00	5.0E-02	1.4E+00	5.0E-05	1.4E-03
Ba-140/La-140	2.26E-03	8.8E-03	2.4E-01	8.8E-06	2.4E-04
Ce-141	8.88E-04	4.2E-04	1.1E-02	4.2E-07	1.1E-05
Ce-144/Pr-144	1.02E-04	6.3E-05	1.7E-03	6.3E-08	1.7E-06
Np-239	1.24E-02	6.8E-03	1.9E-01	6.8E-06	1.9E-04

\* Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.



NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-5b: Normal Operational Non-Volatile Fission Products in  
Reactor Water and Steam**

Isotope*	Decay Constant (per hour)	Water Concentration		Steam Concentration	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
Rb-89	2.74E+00	6.8E-05	1.8E-03	6.8E-08	1.8E-06
Sr-89	5.55E-04	9.2E-08	2.5E-06	9.2E-11	2.5E-09
Sr-90	2.81E-06	4.6E-09	1.2E-07	4.6E-12	1.2E-10
Y-90	2.81E-06	4.6E-09	1.2E-07	4.6E-12	1.2E-10
Sr-91	7.31E-02	6.0E-05	1.6E-03	6.0E-08	1.6E-06
Sr-92	2.56E-01	1.4E-04	3.7E-03	1.4E-07	3.7E-06
Y-91	4.93E-04	1.2E-06	3.3E-05	1.2E-09	3.3E-08
Y-92	1.96E-01	4.1E-05	1.1E-03	4.1E-08	1.1E-06
Y-93	6.80E-02	4.3E-06	1.2E-04	4.3E-09	1.2E-07
Zr-95/Nb-95	4.41E-04	2.5E-06	6.9E-05	6.9E-08	2.5E-09
Mo-99/Tc-99m	1.05E-02	1.2E-05	3.3E-04	3.3E-07	1.2E-08
Ru-103/Rh-103m	7.29E-04	6.1E-07	1.7E-05	1.7E-08	6.1E-10
Ru-106/Rh-106	7.83E-05	9.2E-08	2.5E-06	2.5E-09	9.2E-11
Te-129m	8.65E-04	1.2E-06	3.3E-05	3.3E-08	1.2E-09
Te-131m	2.31E-02	1.6E-06	4.4E-05	4.4E-08	1.6E-09
Te-132	8.89E-03	3.0E-07	8.0E-06	8.0E-09	3.0E-10
Cs-134	3.84E-05	8.3E-07	2.2E-05	2.2E-08	8.3E-10
Cs-136	2.22E-03	6.5E-07	1.8E-05	1.8E-08	6.5E-10
Cs-137/Ba-137m	2.63E-06	1.3E-06	3.4E-05	3.4E-08	1.3E-09
Cs-138	1.29E+00	7.3E-05	2.0E-03	2.0E-06	7.3E-08
Ba-140/La-140	2.26E-03	1.3E-05	3.5E-04	3.5E-07	1.3E-08
Ce-141	8.88E-04	6.1E-07	1.7E-05	1.7E-08	6.1E-10
Ce-144/Pr-144	1.02E-04	9.2E-08	2.5E-06	2.5E-09	9.2E-11
Np-239	1.24E-02	1.0E-05	2.7E-04	2.7E-07	1.0E-08

\* Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

**Table 11.1-6: N16 Concentrations in Reactor Water and Steam\*\*\***

Isotope	Half-Life	Steam Concentration**		Reactor Water Concentration*	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
N-16	7.1 sec	9.25	250	1.78	48

\* During operation with hydrogen water chemistry, increase this value by a factor of five.

\*\* Valid at core exit.

\*\*\* Normal operational concentrations are the same as design basis concentrations.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.1-7: Activated Corrosion Products in Reactor Water and Steam\***

Isotope	Decay Constant (per hour)	Water Concentrations		Steam Concentrations	
		(MBq/gm)	( $\mu$ Ci/gm)	(MBq/gm)	( $\mu$ Ci/gm)
Na-24	4.63E-02	3.2E-05	8.6E-04	3.2E-08	8.6E-07
P-32	2.02E-03	1.2E-06	3.3E-05	1.2E-09	3.3E-08
Cr-51	1.04E-03	3.0E-05	8.0E-04	3.0E-08	8.0E-07
Mn-54	9.53E-05	1.5E-05	4.0E-04	1.5E-08	4.0E-07
Mn-56	2.69E-01	6.4E-05	1.7E-03	6.4E-08	1.7E-06
Fe-55	3.04E-05	3.1E-05	8.3E-04	3.1E-08	8.3E-07
Fe-59	6.33E-04	8.0E-06	2.2E-04	8.0E-09	2.2E-07
Co-58	4.05E-04	7.1E-06	1.9E-04	7.1E-09	1.9E-07
Co-60	1.50E-05	1.4E-05	3.7E-04	1.4E-08	3.7E-07
Ni-63	7.90E-07	3.1E-08	8.3E-07	3.1E-11	8.3E-10
Cu-64	5.42E-02	1.5E-04	4.1E-03	1.5E-07	4.1E-06
Zn-65	1.18E-04	6.4E-05	1.7E-03	6.4E-08	1.7E-06
Ag-110m	1.16E-04	3.1E-08	8.3E-07	3.1E-11	8.3E-10
W-187	2.90E-02	7.2E-06	2.0E-04	7.2E-09	2.0E-07

\* Normal operational concentrations are the same as the design basis concentrations.

## **11.2 Systems for Management of Liquid Radioactive Waste**

### **11.2.1 System and Equipment Functions**

The purpose of the BWRX-300 LWM is to collect liquid radioactive waste generated as the result of normal operation, throughout the plant from the Equipment and Floor Drain System (EFS), separate and filter the waste, then return the filtered water to the Condensate Storage Tank (CST) for plant reuse.

#### **11.2.1.1 Normal Functions (Non-Safety Category)**

The LWM perform the following Non-Safety Category functions during normal conditions:

- The LWM collects potential radioactive liquid waste from plant areas via the EFS for processing and filtering, returning condensate quality water to the CST.
- The LWM also serves to transfer, hold, and filter the Reactor Well Pool volume of water when the reactor head bolts are required to be loosened or tightened during an outage. The Refueling Water Storage Tank (RWST) is used for this purpose.
- The LWM via the CST provides water during outages for under vessel washdown.

#### **11.2.1.2 Normal Functions (Safety Category)**

LWM is designed to perform the following Safety Category functions during normal conditions:

- The LWM provides Containment Isolation Valves (CIVs) on piping that penetrates the containment boundary. One manual, locked closed valve is provided on each side of the containment wall for isolation. The valves are locked closed during normal operations.
- The CST provides an alternate source of water for the Control Rod Drive (CRD) System if the Condensate and Feedwater System (CFS) is not available.

#### **11.2.1.3 Off-Normal Functions (Non-Safety Category)**

The LWM does not perform any non-safety category functions during off-normal conditions.

#### **11.2.1.4 Off-Normal Functions (Safety Category)**

The LWM has the capability to function under off-normal conditions providing a means to replenish the SFP following a Design Basis Accident (DBA). Additionally, the CST provides continuous water to purge the CRD when the CFS is not available.

### **11.2.2 Safety Design Bases**

The CIVs on piping that penetrates the containment boundary provide containment isolation in accordance with CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 11.2-1). These valves are classified Safety Class (SC) 1 due to their Defense Line 3 Safety Category 1 Fundamental Safety Function (FSF) of maintaining containment during and following DBAs. The containment penetration piping and associated valves portion of LWM are classified as Seismic Category A as described in Chapter 3, Subsection 3.2.3.

LWM SSC other than the CIVs are classified SC3 for their Defense Line 2 role in confinement of radioactive materials during normal and off-normal operating conditions.

LWM is designed such that personnel exposures during system operation and maintenance are ALARA per CNSC REGDOC-2.5.2 Section 8.11 (Reference 11.2-1).

The LWM has the capability to process the maximum anticipated quantities of liquid waste without impairing the operation or availability of the plant during both normal and anticipated operational

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

occurrence conditions, satisfying the requirements of CNSC REGDOC-2.5.2 Section 8.11.1 (Reference 11.2-1) and CNSC REGDOC-2.11 (Reference 11.2-2).

In the LWM, radioactive contaminants are removed, and the bulk of the liquid is purified and either returned to the CST or discharged to the environment if the plant's overall water inventory does not permit recycling. The BWRX-300 is designed such that under normal conditions the need to release liquid effluent to the environment is minimized.

Discharges to the environment are monitored by the Process Radiation Monitoring Subsystem (PRM) as described in Section 11.5. The discharge line is automatically isolated upon receiving a signal from PRM to ensure that liquid releases to the environment are within prescribed limits per CNSC REGDOC-2.5.2, Section 8.11.1 (Reference 11.2-1), and that the release of radioactive liquid effluents is below the action limits calculated based on CSA-N288.1 (Reference 11.2-3) and CSA-N288.5 (Reference 11.2-4) and within source dose limits for effluent release cited in SOR/2000-203 (Reference 11.2-5).

### **11.2.3 Description**

The LWM consists of collection tanks, sample tanks, filtration skids, RWST, CST, and associated pumps, piping, and instrumentation. The LWM is illustrated in Figures 11.2-1 to 11.2-2.

#### **11.2.3.1 Equipment, Arrangement, and Location**

The collection tanks, sample tanks and their associated pumps are located in the RWB.

The LWM filtration skids are also located in the RWB. The filtration skids are equipped with shielding to minimize exposure to personnel during operation and routine maintenance.

The RWST and the tank discharge pumps are located in the TB. The Reactor Well Pool filter and the drain pumps are located in the RB.

The CST and associated pumps and valves are located in the TB. CST distribution piping is located throughout the Turbine Building, the RWB and RB.

#### **11.2.3.2 Major Components Description**

##### **11.2.3.2.1 Tanks**

Tanks are sized to accommodate the expected volumes of waste generated in the upstream systems that feed liquid waste into the LWM for processing.

Materials are chosen for the tanks for protection of the tanks from abrasion, wet/dry cycles, freeze/thaw cycles, chemicals, and corrosion. The tanks are provided with mixing eductors. These eductors provide for mixing of the tank's contents to maintain a representative, homogenous fluid sample, and to keep solids in suspension for removal during transfer of the tank's contents. The capability exists to sample all Radwaste tanks. LWM tanks are vented into the Heating, Ventilation and Air Conditioning (HVAC) system of the Radwaste Building. The vents are also large enough to accommodate the airflow associated with pumping down the tanks at a maximum flowrate. To prevent the uncontrolled transport of contamination through those systems, the vents are not hard piped to the HVAC system.

#### **Collection and Sample Tanks**

The collection and sample tanks are cylindrical stainless steel atmospheric tanks sized to accommodate the expected waste volumes generated during normal operation, anticipated operational occurrences, and outage water movement requirements.

### **Refueling Water Storage Tank**

The RWST is a cylindrical stainless steel atmospheric tank sized to accommodate the Reactor Well Pool volume.

### **Condensate Storage Tank**

The CST is a cylindrical stainless steel atmospheric tank located in the TB ground floor. The CST is designed with a reserved capacity to support CRD functions.

### **Pumps**

The LWM process pumps are constructed of materials suitable for their intended service. In the RWB, the pumps are located underneath the corresponding tank. Each tank has a dedicated pump. Pumps are cross tied for flexibility of operation.

### **Filtration Skids**

The filtration skids are designed to process radioactive liquid waste to condensate water quality requirements for plant reuse. The skids utilize filters to remove suspended solids and ion exchangers for removing dissolved ionic compounds. Additional components to increase water quality may also be included on the filtration skids. Exhausted resins and solid wastes are collected into the holdup tanks of the SWM for processing.

### **Piping and Valves**

The containment penetration piping and isolation valves are SC1 and are designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III Class 2 (Reference 11.2-6) and Seismic Category A. All other piping is SC3 and is designed in accordance with the requirements of ASME B31.3 (Reference 11.2-7). LWM piping material is austenitic stainless steel to minimize corrosion. The LWM piping is butt-welded to prevent crud traps. Drainage piping is designed such that the waste from the collection point to the EFS flows by gravity.

#### **11.2.4 Materials**

Material and equipment selection for the system components is based on a planned 60-year design life, with appropriate provisions for maintenance and replacement. Piping and tanks are stainless steel for corrosion resistance. Components which may require periodic replacement and/or maintenance prior to the end of plant life include, but are not limited to, electrical and electronics, O-rings, bearings, gaskets and seals, lubricants, valve disks, seats, and packing.

#### **11.2.5 Interfaces with Other Equipment or Systems**

Refer to Table 11.2-1 for a listing and brief description of the LWM interfaces with other systems.

#### **11.2.6 System and Equipment Operation**

##### **11.2.6.1 Normal Operation**

The LWM normally operates on a batch basis. Liquid radioactive waste is collected into the collection tanks located in the RWB via the EFS. The Shutdown Cooling System (SDC) and the CUW overboarding flow are also routed to the LWM tanks. The EFS, SDC and CUW are described in Chapter 9A.

The liquid from the collection tanks is piped to the filtration skids for batch processing. Each batch is sampled to determine the concentrations of suspended solids and other contaminants. The analysis is completed within the laboratory of the RWB. Processed liquid waste that meets effluent criteria for condensate quality, is pumped to the CST for reuse within the plant or discharged to the environment if the plant's overall water inventory does not permit recycling.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

Discharges to the environment are monitored by PRM and are automatically isolated upon receiving a signal from PRM (See Section 11.5). A recycle line from the sample tanks allows the sampled effluents that do not meet water quality requirements to be reprocessed.

The LWM is equipped with filtration skids for processing liquid waste in batches. Skid-mounted processing equipment along with cross-connections and bypasses are provided to enable efficient waste processing capability and flexibility of operation.

Filters are backwashed periodically to maintain efficiency and resin is replaced when depleted. Backwash waste and spent resin are discharged to the Solid Waste Management System.

During normal operation, when the Main Condenser Hotwell reaches low water level, the CST provides makeup water to the Hotwell.

#### **11.2.6.2 Off-Normal Operation**

The LWM has the capability to function under off-normal conditions providing a means to replenish the SFP following a DBA. Additionally, the CST provides continuous water to purge the Control Rod Drive system when the CFS is not available.

#### **11.2.6.3 Shutdown Conditions**

During shutdown, the Reactor Pressure vessel (RPV) temperature will decrease and cause the water volume in the RPV to shrink. The CST provides the additional makeup water to maintain the normal water level in the RPV.

During refueling outage, when the CFS is not available, the CST provides a continuous purge flow to the CRD system. During leaktest, the CST provides the water to flood the RPV.

The RWST is used to hold the Reactor Well Pool volume when reactor head bolts need to be loosened or tightened during an outage. The piping between the Reactor Well Pool and RWST contains a filter to maintain water purity during movement activities. Water is sampled to ensure water quality meets requirements before being pumped back into the Reactor Well Pool.

During the shutdown/refueling/startup mode, the increases in plant water demand may require use of a supplemental water. That makeup water is fed to the CST by the Water, Gas, and Chemical Pads (WGC) system.

#### **11.2.7 Instrumentation and Control**

The LWM provides the instrumentation to control and monitor (indicate and alarm) system operation.

##### **11.2.7.1 Instrumentation**

The LWM is operated and monitored from the Main Control Room (MCR). Major system parameters, i.e., tank levels, process flow rates, filters, and filtration skids effluent conductivity, etc., are indicated and alarmed as required to provide operational information and performance assessment. PRM (discussed in Section 11.5) continuously monitors the discharge of radioactivity to the environs.

##### **11.2.7.2 Control**

The LWM is controlled and monitored from the MCR.

The CST makeup water supply control valve of the WGC system modulates on the basis of level transmitters monitoring water level in the CST to control the water level in the tank. It also can be initiated manually from the MCR.

The condensate transfer pumps have the capability to be initiated manually locally and from the MCR. One of the pumps is normally operated in standby mode. The standby pump automatically starts when the running pump trips or when system flow demand exceeds the capacity of the operating pump.

The Main Condenser Hotwell is provided with two flow paths for Hotwell makeup, a normal flow path and an emergency flow path. Both flow paths are equipped with control valves that modulate on the basis of water level in the Main Condenser Hotwell to control the level of water in the Hotwell. The emergency valve modulates when the normal path cannot meet demands. Both valves also can be initiated manually from the MCR.

The flow of the liquid waste to the collection tanks is controlled by the EFS sump redundant level transmitters. Level signals can be used to start and stop the sump pumps based on high and low settings. A high LWM tank level initiates a high-level alarm in the MCR for the affected tank. Closure of the tank's inlet valves can be controlled remotely from the MCR to prevent overflow.

The environmental discharge controls consist of a flow control valve and two isolation valves in series that automatically close upon receiving a signal from the PRM and initiates a high radiation alarm in the MCR to ensure that release of radioactive liquid effluents meets the limits prescribed in the Radiation Protection Regulations (Reference 11.2-5), action levels and operational targets for releases in accordance with CNSC REGDOC-2.5.2, Section 8.11 (Reference 11.2-1), and per guidance in CSA N288.8 (Reference 11.2-8).

#### **11.2.8 Monitoring, Inspection, Testing, and Maintenance**

Provisions are made for periodic inspection of major components to ensure capability and integrity of the systems. The system design provides adequate equipment removal paths and personnel access for replacement per General Arrangement Drawings. Adequate space is provided in the RWB for maintenance of the pumps and valves and instruments.

#### **11.2.9 Radiological Aspects**

##### **Normal Operating Conditions**

Chapter 20, Section 20.14 summarizes analysis performed to determine the annual average liquid effluent releases from the LWM for normal operating conditions (including Anticipated Operational Occurrences (AOO)), based on normal operation source terms in the reactor coolant and steam described in Subsection 11.1.2. The results are used as an input to calculation of public dose show that the BWRX-300 is below regulatory limits.

As was discussed earlier, accidental discharge and discharge above regulatory limits is prevented through detection by PRM and alarm of abnormal conditions and by administrative controls. The LWM is designed to automatically stop liquid discharge from the sample tanks upon receiving a signal from the PRM. Offsite radiation exposures on an annual average basis are within the limits of CSA-N288.1 (Reference 11.2-3) and CSA-N288.5 (Reference 11.2-4).

##### **Leaks from Liquid Waste Management System Components**

To prevent tank leakage from exiting the building, the collection and sample tanks are enclosed within a concrete wall barrier. In the case that a tank failure occurs, the concrete wall containment area will prevent LWM water from exiting the RWB.

The overflow pipes of the tanks that hold radioactive liquid are designed to run to the appropriate radioactive drain collection system, and the vents are designed to run to the building's ventilation exhaust system, but neither are hard connected to their respective drain or vent systems to prevent the uncontrolled transport of contamination through those systems.



### **11.2.10 Performance and Safety Evaluation**

The two manual isolation CIVs are locked closed, only opened during refueling Mode 6 and are administratively controlled.

The performance of the LWM to minimize liquid effluent releases during normal operation to within prescribed limits and action levels is described in the analysis in Chapter 20, Section 20.14.

### **11.2.11 References**

- 11.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 11.2-2 CNSC Regulatory Document REGDOC-2.11, "Framework for Radioactive Waste Management and Decommissioning in Canada."
- 11.2-3 CSA N288.1-20, "Guidelines for modeling radionuclide environmental transport, fate and exposure associated with normal operations of nuclear facilities," CSA Group.
- 11.2-4 CSA N288.5-11, "Effluent Monitoring Programs At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 11.2-5 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 11.2-6 ASME BPVC, "Section III- Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers.
- 11.2-7 ASME B31.3, "Process Piping," American Society of Mechanical Engineers.
- 11.2-8 CSA N288.8, "Establishing and implementing action levels for releases to the environment from nuclear facilities," CSA Group.
- 11.2-9 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.2-1: System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-safety Instrumentation and Control System	Provides non-safety I&C control, logics, and instrumentation.	LWM Equipment
Process Radiation Monitoring Subsystem (PRM)	LWM isolates liquid discharge upon receiving a trip signal from PRM.	LWM Equipment
Control Rod Drive System (CRD)	CST provides continuous water to purge CRD when the CFS is not available.	LWM Equipment
Shutdown Cooling System (SDC)	SDC provides a reactor water reject path for cleanup and over boarding to LWM when normal flow paths (e.g., CUW) are not in-service.	LWM Equipment
Reactor Water Cleanup (CUW)	LWM receives over boarding flow from CUW to the RWST / Collection Tanks.	LWM Equipment
Fuel Pool Cooling and Cleanup System (FPC)	CST provides water supply to FPC for use.	LWM Equipment
Control Panel System (CPS)	Local Control Panels provided for operation of skids.	Filtering Skid Controls
Solid Waste Management System (SWM)	Oily and solid wastes are collected and pumped to SWM for disposal.	LWM Equipment
Condensate and Feedwater Heating System (CFS)	CFS rejects water to the CST to control Hotwell level.	LWM Equipment
Condensate Filters and Demineralizer System (CFD)	CST provides water supply to the CFD for use.	LWM Equipment
Main Condenser and Auxiliaries (MCA)	CST accepts water from MCA and provides Hotwell makeup.	LWM Equipment
Circulating Water Equipment (CWS)	CWS accepts LWM effluent discharge to environment.	LWM Solenoid Valves
Plant Pneumatics System (PPS)	PPS provides air needed to air operate valves and dewatering pump for control of the system.	LWM AOV's
Non-safety Electrical Distribution System	Provides Non-safety power to LWM equipment.	LWM Equipment
Primary Containment System (PCS)	LWM CST provides water to inside containment for outages.	PCS Locked closed valve
Heating, Ventilation and Cooling System	Tank vents are discharged into the HVAC system of the Radwaste Building.	LWM Tanks
Equipment and Floor Drain System (EFS)	EFS collects and pumps radioactive liquid wastes to the collection tanks of LWM.	LWM Equipment
Water, Gas, and Chemicals Pads System (WGC)	The WGC provides temporary makeup water during the shutdown/refueling/startup mode when required by the increases in plant water demand.	CST

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

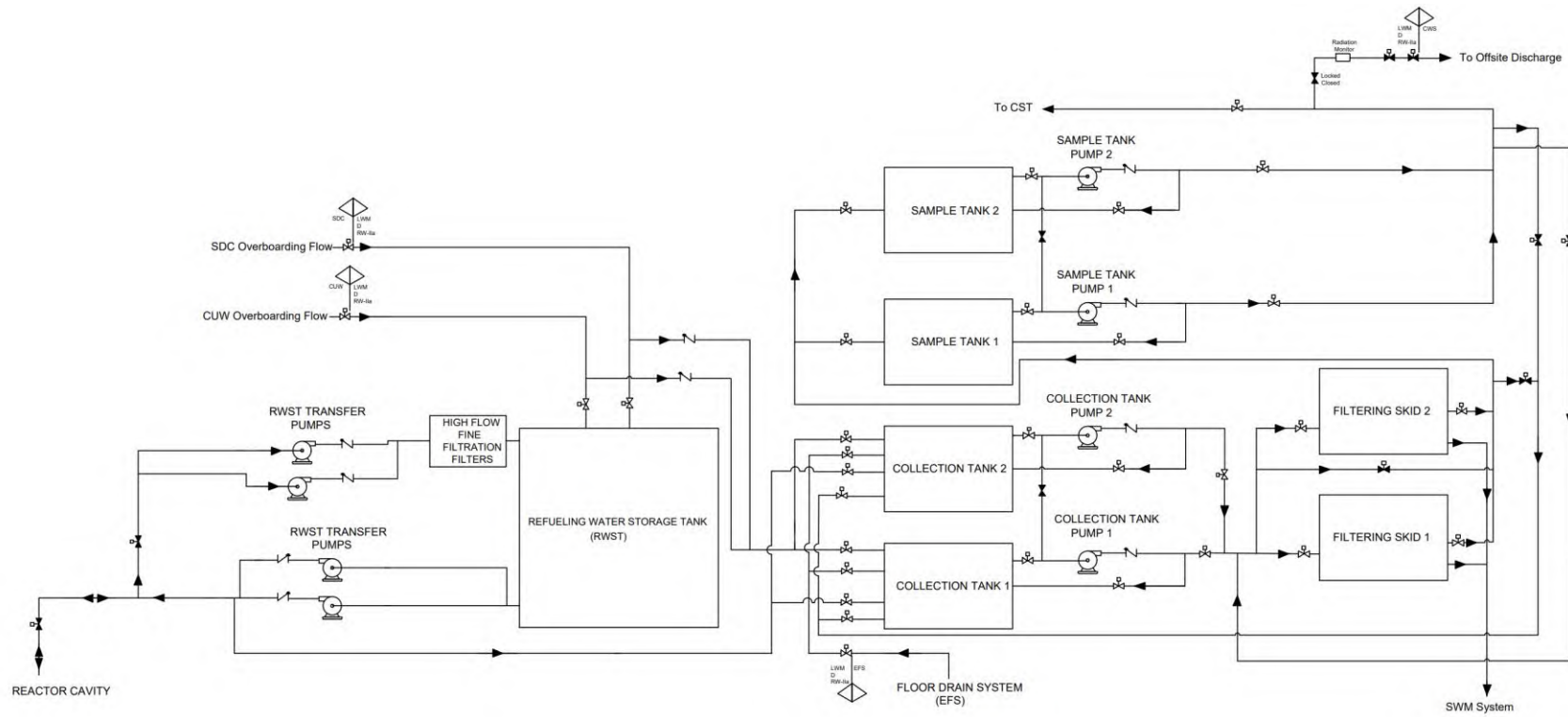


Figure 11.2-1: LWM Simplified Flow Diagram

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

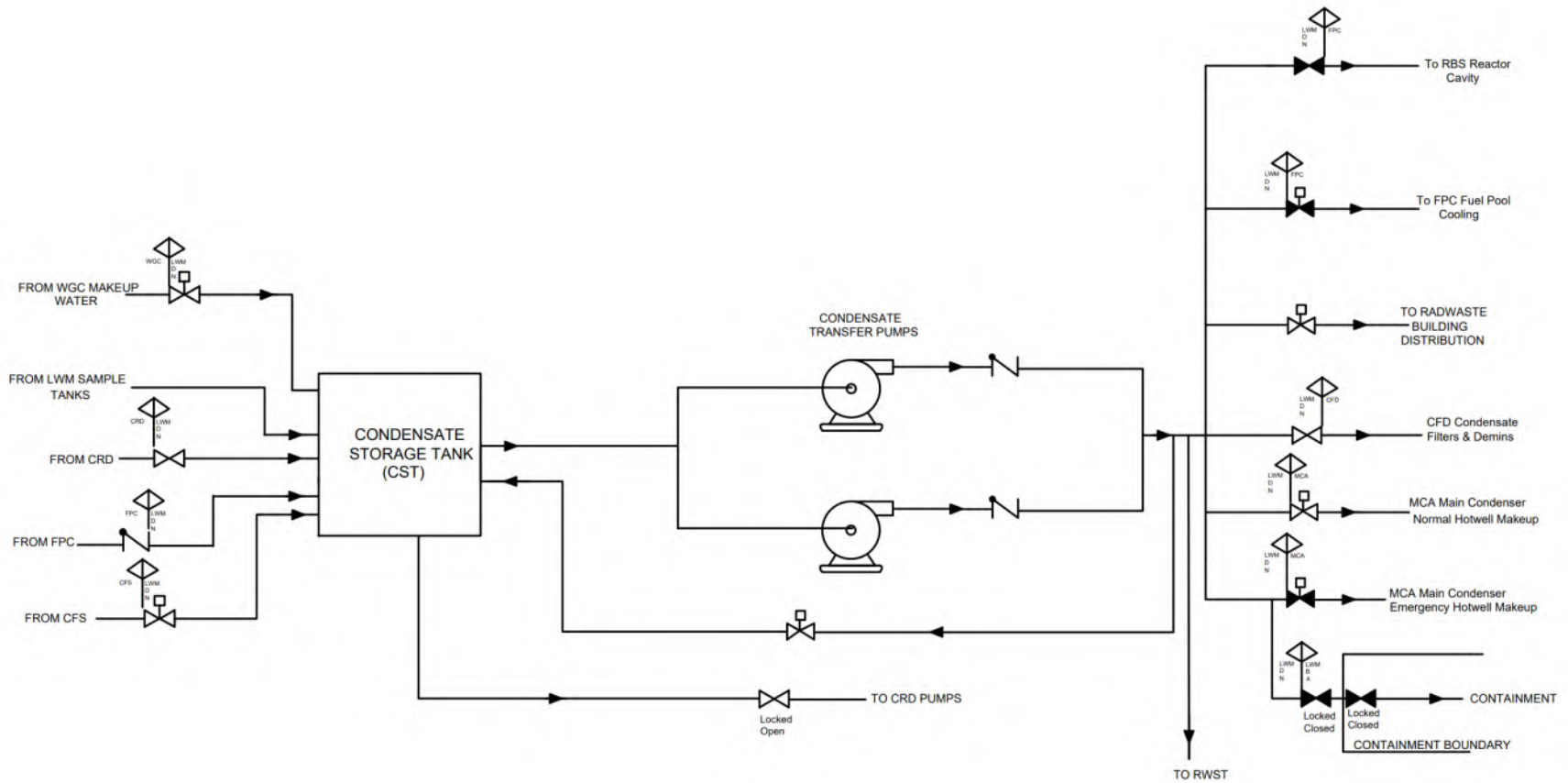


Figure 11.2-2: Condensate Storage Tank Users Simplified Flow Diagram

### **11.3 System for Management of Gaseous Radioactive Waste**

#### **11.3.1 System and Equipment Functions**

The objective of the Offgas System (OGS) is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons in unrestricted areas to radioactive gaseous effluents ALARA per CNSC REGDOC-2.5.2, Section 8.11.2 (Reference 11.3-1).

#### **11.3.2 Normal Functions (Non-Safety Category)**

The OGS does not perform any non-safety category functions during normal conditions.

#### **11.3.3 Normal Functions (Safety Category)**

The OGS performs the following Safety Category 3 FSF:

- Recombine hydrogen and oxygen back into water vapor to route back to the main condenser to reduce the explosion potential in downstream process components
- Provide a hold up via carbon adsorption for fission product gases to reduce the radioactive dose in the effluent released from the plant to the environment to levels below regulatory dosage requirements

#### **11.3.4 Off-Normal Functions (Non-Safety Category)**

The OGS does not perform any Non-Safety Category functions during off-normal conditions.

#### **11.3.5 Off-Normal Functions (Safety Category)**

The Charcoal Vault is designed such that the first charcoal bed in the series may be bypassed, referred to as the "Guard Bed". In the event that condensation occurs in the first charcoal bed, the bed may be bypassed, allowing for effluent to flow through the remaining three Charcoal Beds without carrying moisture from the first charcoal bed, and maintaining holdup for the Kkryptons and xenons. Acceptable holdup times are provided by the remaining three beds such that normal operation may continue until maintenance is able to restore the Guard Bed to normal operating conditions

#### **11.3.6 Safety Design Bases**

The OGS is designed to meet CNSC REGDOC-2.5.2, Section 8.11.3 requirements for control of radioactive contaminants to the environment and for maintaining airborne radioactive contaminants within prescribed limits and ALARA during reactor power operation. The OGS is an SC3 system due to its support of Defense Line 2 FSF fuel cooling and confinement of radioactive materials during normal operation.

The portion of the OGS within the TB is seismically categorized as Non-Seismic, while the portion of the OGS housed within the RWB is classified as RW-IIa.

#### **11.3.7 Description**

##### **11.3.7.1 Summary System Description**

The OGS provides for holdup and decay of radioactive gases in the offgas from the Main Condenser and Auxiliaries system.

The OGS minimizes and controls the release of radioactive material into the atmosphere by delaying and filtering the offgas process steam containing the radioactive isotopes of krypton, xenon, iodine, hydrogen, nitrogen, and oxygen sufficiently to achieve adequate decay before discharge from the plant.

The OGS is comprised of the following components shown in Figure 11.3-1:

- "3-in-1" Offgas Recombiner unit, including Offgas Preheater section, Offgas Catalytic Recombiner section, and Offgas Condenser section
- Cooler Condenser and Moisture Separator
- Refrigeration Dryers (two)
- Gas Analyzer
- Offgas Reheater
- Charcoal Adsorber Vault with Charcoal Adsorber Beds (four)
- Charcoal Vault HVAC equipment
- Offgas HEPA Filter
- Piping and Valves
- Instrumentation and Control equipment

#### **11.3.7.2 Equipment, Arrangement, and Location**

Equipment for the OGS is located within the TB and the RWB. The primary effluent line of the OGS begins at the valve interface with the Main Condenser and Auxiliaries system at the exit of the Steam Jet Air Ejectors (SJAES) within the TB.

The following components of the OGS are housed in the TB:

- Entrance process piping
- Offgas Recombiner
- Cooler Condenser and Moisture Separator
- Refrigeration Dryers (two)
- Gas Analyzer

After exiting the Refrigeration Dryers, OGS process piping carries the offgas effluent from the TB to the RWB.

The following components of the OGS are housed in the RWB:

- Offgas Reheater
- Charcoal Vault, including four Charcoal Adsorber Tanks
- Offgas HEPA Filter

The OGS process ends with the transfer of the offgas effluent to the Continuous Air Plenum (CAP) of the Heating Ventilation and Cooling System prior to final release.

#### **11.3.7.3 Structures, Systems and Components**

Figure 11.3-2 provides a simplified single line diagram for the OGS. The subsystems, components and their functions are described in the following subsections.

##### ***11.3.7.3.1 Offgas Recombiner***

The Offgas Recombiner is a "3-in-1" vessel consisting of a Preheater section, Catalytic Recombiner section, and Offgas Condenser section. The purpose of the Offgas Recombiner is

to facilitate the catalytic recombination of radiolytic oxygen and hydrogen into water, which is ultimately condensed via a series of downstream cooling equipment and returned to the main condenser.

The Preheater Section is a tube-and-shell heat exchanger. The purpose of the Preheater section is to raise the temperature of the offgas so that the water content reaches a superheated state, ensuring that it is fully boiled and gaseous prior to entering the Catalytic Recombiner section. Effluent offgas from the SJAEs, consisting of steam, air, hydrogen, oxygen, and other trace radioactive gases, enters the Catalytic Recombiner section through the Preheater section and flows through the shell side. Steam from the Main Steam line flows through the tube side and is used to heat the offgas. The Main Steam is condensed from a saturated vapor state to a saturated liquid state. The condensed Main Steam water is drained and returned to the main condenser, while the superheated offgas flows to the Catalytic Recombiner section. The Preheater section contains a pressure relief line to the main condenser, temperature element, and condensed steam drain line to the main condenser with a steam trap.

The Catalytic Recombiner section uses a precious metal catalyst bed to cause the recombination of radiolytic oxygen and hydrogen gases into water. Temperature and heating elements maintain the catalyst environment to acceptable temperature and moisture levels prior and during operation. For maintenance, normally-closed drains are included. Offgas flows from the Catalytic Recombiner section into the Offgas Condenser section.

The Offgas Condenser section is a tube-and-shell heat exchanger. This component provides the first stage of condensation of both the dilution steam in the effluent mix along with the additional water vapor resulting from the recombination of the radiolytic gases. The offgas effluent enters the shell side of the heat exchanger. Condensate water from the CFS enters the tube side of the heat exchanger and is used to cool the offgas effluent. Condensate water passes through the Condenser Section and returns to the CFS. The condensed liquid water from the offgas is drained to the main condenser. The remaining gaseous offgas effluent flows forward to the Cooler Condenser. The Offgas Condenser section includes a pressure relief line to the main condenser, flow element to monitor offgas condensate flowing to the main condenser, and temperature elements to monitor the Feedwater Condensate and gaseous offgas leaving the Offgas Recombiner.

#### ***11.3.7.3.2 Cooler Condenser***

The Cooler Condenser Section is a tube-and-shell heat exchanger. This component provides the second stage of condensation of the offgas. The offgas effluent enters the shell side of the heat exchanger. Chilled water from the Chilled Water Equipment (CWE) enters the tube side of the heat exchanger and is used to cool the offgas effluent. Chilled water passes through the Cooler Condenser and returns to the CWE. As most moisture is condensed in the Offgas Recombiner Condenser section, the condensed liquid water from the Cooler Condenser is drained to local floor drains. The remaining gaseous offgas effluent flows forward to the Moisture Separator.

#### ***11.3.7.3.3 Moisture Separator***

The Moisture Separator is downstream of the Cooler Condenser. This is an in-line tank with a wire mesh or chevron screen-type media which serves to divert any moisture droplets remaining in the effluent stream to a drain.

#### ***11.3.7.3.4 Refrigeration Dryers***

The Refrigeration Dryers provide the final stage of cooling and condensation removal of the offgas effluent. Two dryers are provided in parallel, though only one is active at any given time. This is

to ensure that operation of the OGS may continue if one dryer needs maintenance. Condensed water is drained to local floor drains.

#### ***11.3.7.3.5 Offgas Reheater***

Downstream of the Dryers is the Reheater. This serves the function to raise the effluent flow stream temperature from the refrigeration dryers to the higher temperature of the charcoal in the Charcoal Vault to prevent moisture in the flow stream from condensing onto the charcoal due to the temperature difference between the flow stream and the charcoal. This component is composed of a series of electrical resistance heaters mounted onto the exterior of the effluent pipe just upstream from the Charcoal Vault.

#### ***11.3.7.3.6 Charcoal Adsorber Vault and Tanks***

The OGS Charcoal Vault is a shielded and isolated room in which radioactive isotopes are contained and held up to allow for decay prior to release. Charcoal Vault pressure is maintained slightly lower than atmospheric pressure relative to the general area pressure to mitigate the risk of noble gas leaks. The Charcoal Vault HVAC system maintains the ambient temperature of the vault at a constant temperature. Vault temperature is continuously monitored, and the HVAC system has a redundant train of heating and cooling equipment.

The Charcoal Beds consist of four vertical tanks of granulated activated charcoal designed to hold up the targeted krypton and xenon isotopes so as to reduce released effluent activity. These tanks (and their contained charcoal volume) are designed to provide the required holdup for targeted noble gas isotopes to meet site requirements for release below action levels calculated per CSA N288.8 (Reference 11.3-2) ensuring prescribed limits for effluent releases cited in SOR/2000-203 (Reference 11.3-3). This is done through the process of adsorption, by which the atoms of the krypton and xenon gases are held (by Van der Waal-type forces) to the surfaces of the charcoal granules long enough to delay their migration downstream. Multiple temperature elements monitor the temperatures within each tank. A Carbon Monoxide (CO) monitor is provided to detect combustion of the charcoal, and nitrogen may be injected into the vault piping to extinguish the fire if necessary.

The first charcoal bed, known as the Guard Bed, is configured to be bypassed if a problem manifests in it (e.g., high moisture or potential combustion issue) such that the system could operate on the latter three beds (for short periods of time). As this is the first bed to process the effluent, it is most susceptible to any issues caused by a malfunction of the upstream equipment. Bypassing the Guard Bed is acceptable for short periods of time until any issue with that bed (and associated equipment) can be remedied.

The offgas effluent flows through the Guard Bed, then through the other three beds, before moving forward to the final offgas HEPA Filter for release. The entire Charcoal Vault may also be bypassed during offgas startup or off-normal operation.

#### ***11.3.7.3.7 Offgas HEPA Filter***

Downstream from the Charcoal Beds is an in-line HEPA filter. This is a small filter contained in a tank which serves to pick up any fine pieces of charcoal and any other particulates which may pass downstream of the Charcoal Beds immediately prior to offgas release. Differential pressure across the filter is monitored to detect whether a buildup of particulate within the filter is occurring.

#### ***11.3.7.3.8 Gas Analysis Equipment***

Hydrogen and oxygen concentration and moisture content are monitored at the outlet of the Dryer. The annunciators of high hydrogen concentration and high/low oxygen concentration alarm are provided at the MCR, and also serve as an input to the Hydrogen Water Chemistry system to adjust the injection of hydrogen and oxygen as needed. CO monitoring is provided at the outlet



of the Charcoal Vault to detect for fire. Radioactivity level in the OGS is monitored at the outlet of the dryers and just prior to release, and high radioactivity level trips an alarm in the MCR.

#### **11.3.8 Materials**

Material and equipment selection for the OGS are based on a 60-year design life, with appropriate provisions for maintenance and replacement.

#### **11.3.9 Interfaces with Other Equipment or Systems**

The interface between the OGS and other systems is listed in Table 11.3-1.

#### **11.3.10 System and Equipment Operation**

##### **11.3.10.1 Normal Operation**

The primary functions of the OGS are most applicable when the reactor is operating at high power. Production of radiolytic H<sub>2</sub> and O<sub>2</sub>, and of radioactive noble gas isotopes, are minimal outside of Reactor Power Operation mode, and as such, the OGS is not required to be in operation during Reactor Startup, Shutdown, and Refueling modes. The OGS System Modes coinciding with Plant Operating Modes are listed in Table 11.3-2 and described below.

##### ***11.3.10.1.1 Initial Configuration (Pre-Startup) Mode C (Secured))***

The OGS is secured (inactive) during initial startup operations (0-5% reactor power), all shutdown modes, and refueling. Because the reactor is at very low or no power during these operating modes, and because these modes occur for a relatively short period of time, the OGS is not required to process radiolytic gases in the effluent stream, and so the OGS is bypassed. The Mechanical Vacuum Pump is active and drawing main condenser vacuum, and valves controlling effluent flow to the OGS are closed.

##### ***11.3.10.1.2 System Startup (Mode B)***

The Startup Operation Mode of the OGS occurs between the time that the SJAEs are placed into service (at around 5% reactor power) as the reactor power rises, until reactor power has reached 10% power. At the beginning of startup, the Catalytic Recombiner section heaters are activated to ensure a sufficiently high temperature is present when the offgas effluent begins to flow through the OGS. This mode begins when the Mechanical Vacuum Pump is secured (i.e., no longer drawing main condenser vacuum). Radiation levels (in the effluent flow stream) are relatively low at this time.

During Startup Operation Mode, the effluent stream flows through a bypass valve around the Charcoal Vault, then releases to the HVAC system. This is to prevent damage to the Charcoal Beds due to the increased air in-leakage rate during startup.

During OGS startup, there is little hydrogen and oxygen to recombine as radiolysis is directly dependent on (and directly proportional to) reactor power to drive the process. As such, there is little temperature rise seen in the Offgas Recombiner during this mode (due to the minimal heat added by the exothermal process of recombination). Likewise, leakage rates of radioactive isotopes (kryptons and xenons) into the offgas effluent are minimal at this stage.

##### ***11.3.10.1.3 Normal Operation (Mode A)***

The Normal Operation Mode of the OGS begins at or just beyond 10% reactor power, where main condenser vacuum is stable, much of the in-leakage has subsided, and higher levels of radiation (in the flow stream) are occurring due to the rise in noble gas release from the fuel in the form of xenons and kryptons. The flow of effluent is routed through all primary system components, including the Offgas Recombiner, Cooler Condenser, Moisture Separator, Refrigeration Dryer, Charcoal Vault and all Charcoal Tanks, and the Filter before being released.

During normal operation, radiolytic gas production varies linearly with thermal power.

During the beginning of the normal Mode of operation, the Catalytic Recombiner temperature steadily rises with reactor power up to 100% reactor power, at which point the Catalytic Recombiner temperature stabilizes.

One dryer is in-service, with the other on standby.

#### **11.3.10.2 Off-Normal Operation**

The Charcoal Vault is designed such that the first charcoal bed in the series (referred to as the "Guard Bed") may be bypassed. If condensation occurs in the first charcoal bed, the bed may be bypassed, allowing for effluent to flow through the remaining three Charcoal Beds preventing moisture passing from the first charcoal bed, and maintaining holdup for the kryptons and xenons. Acceptable holdup times are provided by the remaining three beds such that normal operation may continue until the Guard Bed is restored to normal operating conditions.

#### **11.3.10.3 System Shutdown**

OGS Shutdown operation essentially functions as a reversal of the startup operation. As the reactor powers down, the OGS will continue operating until the main condenser vacuum is switched to the Mechanical Vacuum Pump. When vacuum flow is no longer flowing through the OGS, a final purge flow from the Plant Pneumatic System (PPS) is sent through the OGS to eliminate any remaining gas in the piping. When the plant has reached SHUTDOWN operation mode, the OGS is secured.

#### **11.3.11 Instrumentation and Control**

##### **11.3.11.1 Minimum Offgas Flow Rate Control and Air Injection**

A minimum offgas flowrate is maintained at all times while there is flow through the Offgas Recombiner to support normal function of the Catalytic Recombiner.

Flow detectors continuously measure the minimum offgas flowrate and transmit a control signal upstream to the Main Condenser and Auxiliaries to inject additional air into the effluent flow if this minimum is not met.

##### **11.3.11.2 Instrumentation**

The following are the key instrumentation features of the OGS:

1. Temperature and pressure of noncondensable gases entering the Offgas Recombiner are continuously monitored in the MCR. These measurements are used to alarm on low gas temperature (which may impact the catalytic recombination operation) and high gas pressure (which may exceed the design pressure of the Offgas Recombiner vessel.).
2. Temperature of the offgas effluent stream is measured in the Preheater Section and at different locations within the Catalytic Recombiner section using temperature elements. These measurements are displayed in the MCR to assure that recombination is occurring. Low (Preheater Section) and high/low (Catalytic Recombiner section) temperatures generate alarms in the MCR.
3. Temperature of the offgas effluent stream is measured after the Offgas Condenser section with temperature elements to assure the effluent is cooled sufficiently to remove undesired moisture.
4. Temperature of the offgas effluent stream is measured after the Cooler Condenser with temperature elements to assure the effluent is cooled sufficiently to remove undesired moisture.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

5. Offgas effluent flowrate downstream of the Offgas Recombiner and downstream of the Charcoal Adsorber Vault is continuously monitored to permit monitoring of fission gases from the reactor and calculation of offgas discharge to the vent and will permit calculation of the charcoal adsorber system performance.
6. Radiation levels at two sample points, one just downstream of the Refrigeration Dryers, and one just downstream of the Charcoal Vault, are monitored and are continuously displayed in the MCR. These measurements are used to alarm on high radiation levels.
7. Drain seal loops on the Offgas Condenser section are monitored using level transmitters that alarm in the MCR on low/high water level.
8. A CO monitor downstream of the first Charcoal Adsorber Tank (the Guard Bed) and another CO monitor downstream of the last charcoal adsorber are used to confirm the presence and cessation of charcoal combustion within the vessels.

### **11.3.11.3 Control**

Controls for the OGS are available in the MCR to facilitate system operation. System and component operating status, including the state of any bypasses or manual overrides, are provided at the control console. Manual initiation and shutdown of the OGS is provided from the control room.

OGS parameter indications are provided in the MCR. The indications available included but not limited to: Offgas Recombiner temperatures and flows, Dryer Outlet hydrogen and oxygen content, Charcoal Adsorber bed temperatures, and OGS release effluent radiation and flow rates.

OGS alarms are provided in the MCR based on specific setpoints for the Offgas Recombiner indications, Charcoal Adsorber bed temperatures, Hydrogen and Oxygen levels, and OGS Discharge Radiation levels.

### **11.3.12 Monitoring, Inspection, Testing, and Maintenance**

#### **11.3.12.1 Monitoring**

Regular sampling of the offgas is conducted for isotopic analysis of the offgas to ensure that the required limits for the annual average atmospheric dispersion factor are satisfied. This sample occurs on an established frequency, under normal operating conditions to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The frequency is established based on operating experience and is in addition to other instrumentation that continuously monitors the offgas. The sampling is not required to be performed until radioactive fission gases are present in the Main Condenser Offgas System at significant rates.

Individual component maintenance and inspection requirements will be provided in vendor specification documentation.

The design of the OGS components includes, where appropriate, any special tools or special facilities which are required to disassemble, assemble, or otherwise perform maintenance on those components.

#### **11.3.12.2 Accessibility for Maintenance**

Equipment will not normally be accessible for maintenance during system operation. All equipment is available during the plant outage. The following are exceptions:

1. The redundant refrigeration dryer is accessible for maintenance while inactive.
2. Control valving and hydrogen analyzers are accessible for maintenance during the out-of-service portion of their cycle.

3. Charcoal Vault air HVAC equipment is accessible for maintenance during plant operation.
4. The isolation valves in the nitrogen and air purge lines are accessible for maintenance from outside the Charcoal Vault and when operating the OGS.

#### **11.3.12.3 Surveillance Testing and In-Service Inspection Provisions**

Routine testing of the OGS is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

The system has no requirements for In-Service Inspection.

#### **11.3.13 Radiological Aspects**

##### **Normal Operating Conditions**

Chapter 20, Section 20.13 summarizes analysis performed to determine the annual average gaseous effluent releases (including releases from the OGS) for long-term normal operation (including AOOs).

The OGS minimizes and controls the release of radioactive material into the atmosphere by delaying release of the offgas process stream initially containing radioactive isotopes of krypton, xenon, iodine, nitrogen, and oxygen. This delay, using activated charcoal adsorber beds, is sufficient to achieve adequate decay before the process offgas stream is discharged from the plant.

The gaseous effluent treatment systems are designed to limit the dose to offsite persons from routine station releases to significantly less than the limits in Radiation Protection Regulations (Reference 11.3-3) and below the action levels established per CSA N288.8 (Reference 11.3-2).

Charcoal Vault pressure is maintained slightly lower than atmospheric pressure relative to the general area pressure to mitigate the risk of noble gas leaks from the vault into the wider plant.

##### **OGS Leak or Failure**

The limiting BWRX-300 OGS failure is an inadvertent bypass of the charcoal delay beds. An evaluation was performed, and the resulting public dose consequences are well below the CNSC REGDOC-2.5.2, Section 4.2.1 (Reference 11.3-1) limits as shown in Table 11.3-3.

#### **11.3.14 Performance and Safety Evaluation**

The performance of the OGS to minimize gaseous effluent releases during normal operation to within prescribed limits and action levels is described in the analysis in Chapter 20, Section 20.13.

#### **11.3.15 References**

- 11.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 11.3-2 CSA N288.8, "Establishing and implementing action levels for releases to the environment from nuclear facilities," CSA Group.
- 11.3-3 Government of Canada SOR/2000-203, "Radiation Protection Regulations."

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.3-1: System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Nuclear Boiler System (NBS)	Input: Reactor steam is provided to the OGS to heat the effluent entering the Preheater. Steam enters the OGS piping, enters the tube side of the Preheater is condensed to liquid, and is ultimately ejected to the MCA.	Boundary Air Operated Valve (AOV) is owned by NBS.
SC2 and SC3 Digital Control and Information System (DCIS)	NS DCIS receives/provides signals to S I&C equipment in the OGS.	Throughout the system.
Process Radiation Monitoring Subsystem (PRM)	Output: Offgas effluent are sampled and monitored by PRM downstream of dryers and prior to release to ensure offgas radiation content is below the acceptable release limit.	Release sample flow to Rad Monitoring. Post-dryer monitoring.
Condensate and Feedwater System (CFS)	Output: CFS reactor condensate water flows through the tube side of the Offgas Condenser. Effluent heat is rejected to the CFS condensate. CFS condensate water is returned to the CFS system piping after exiting the Offgas Condenser.	Piping immediately up to the fitting on the Offgas Condenser is owned by CFS. The boundary where condensate water returns from OGS to CFS.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Main Condenser and Auxiliaries (MCA)	Input: Effluent gases (including air in-leakage and noncondensable gases) and dilution steam from the MCA SJAEs flow directly to OGS.	Boundary valve owned by MCA.
	Input: Air Injection. When additional air is required to maintain the minimum required flow of 0.17 sm <sup>3</sup> /min (6 scfm) through the Offgas Recombiner, air may be injected into the effluent line upstream of OGS within the boundaries of MCA.	Boundary valve owned by MCA. Additional air is injected directly into incoming effluent line.
	Output: Condensed water from Offgas Preheater tube side (originating from Main Steam) and from the Offgas Condenser shell side (originating from the dilution steam from the SJAEs) is drained to the main condenser.	Boundary to MCA is at the main condenser itself.
	Output: Flowrate of air into the Offgas Recombiner is monitored to ensure minimum required flow 0.17 sm <sup>3</sup> /min (6 scfm) is met. This flowrate is sent as a control signal to MCA to indicate if air injection is required.	Flow Indicator/transmitter located prior to entering Offgas Preheater.
Chilled Water Equipment (CWE)	Output: Chilled water is used to provide cooling to the Cooler Condenser and to the cooling coils in the Charcoal Vault HVAC system. Chilled water flows through the equipment then returns directly to CWE.	Boundary valves owned by CWE.
Plant Pneumatics System (PPS)	Input: Motive air for AOV valve control provided by the PPS.	At each valve.
	Input: Nitrogen flow to purge Charcoal Tanks during maintenance periods.	N <sub>2</sub> inlet within the Charcoal Vault.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

Interfacing System	Interface Description	Interface Boundary
Hydrogen Water Chemistry	Input: Hydrogen Water Chemistry control.	Hydrogen Water Chemistry may inject additional hydrogen into the effluent flow upstream of OGS, to be recombined with oxygen into water in the Offgas Recombiner.
	Output: hydrogen and oxygen monitoring and control signal.	A Gas Analyzer in OGS will sample the flow downstream of the Refrigeration Dryers to monitor H <sub>2</sub> and O <sub>2</sub> content in the effluent flow, sending a signal to Hydrogen Water Chemistry if the detected gas content is above a set threshold.
SC2 and SC3 Electrical Distribution System	Input: Non-Safety Electrical Distribution System to provide power to non-safety category electrical loads in OGS system.	Interface to be located at local breaker.
Heating Ventilation and Cooling System	Output: Exhaust heat from Offgas equipment (Refrigeration Dryers and Charcoal Vault cooling), offgas effluent release.	TB HVAC, RWB HVAC.
Equipment and Floor Drain System (EFS)	Output: Water condensed from the effluent in the Cooler Condenser, through the Moisture Separator, and the Refrigeration Dryers are drained to EFS.	Cooler Condenser, Moisture Separator, and Refrigeration Dryer drain lines.
Turbine Building Structure	Input: OGS equipment is housed within TB.	TB interior.
Radwaste Building	Input: OGS Charcoal Vault and downstream equipment is housed within the RWB.	RWB interior.

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.3-2: BWRX-300 Operating Modes**

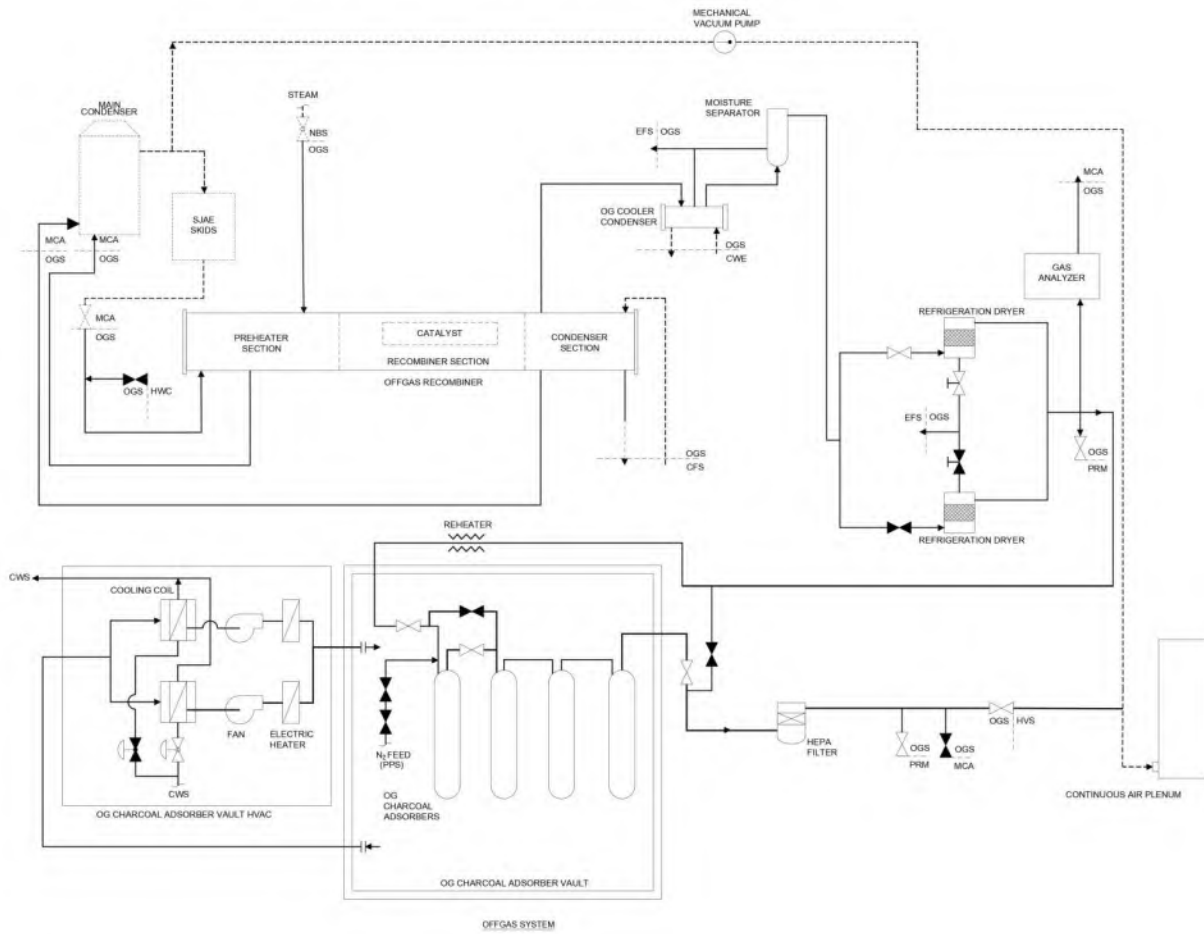
<b>Mode</b>	<b>Title</b>	<b>Reactor Mode Switch Position</b>	<b>OGS Modes</b>
1	POWER OPERATION (10% - 100% Rx Pwr)	RUN	A (Normal Operation Mode)
1a	POWER OPERATION (5% - 10% Rx Pwr)	RUN	B (Startup Operation Mode)
2	STARTUP	REFUEL or STARTUP	C (Secured)
3	HOT SHUTDOWN	SHUTDOWN	C (Secured)
4	STABLE SHUTDOWN	SHUTDOWN	C (Secured)
5	COLD SHUTDOWN	SHUTDOWN	C (Secured)
6	REFUELING	SHUTDOWN or REFUEL	C (Secured)



**Table 11.3-3: BWRX-300 Offgas System Failure Dose Results**

Committed Whole Body Dose	CNSC REGDOC-2.5.2 Dose Limits	
	AOO Limit	DBA Limit
0.32 mSv (32 mrem)	0.5 mSv (50 mrem)	20 mSv (2,000 mrem)

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 11.3-1: OGS Simplified Flow Diagram**

## **11.4 System for Management of Solid Radioactive Waste**

### **11.4.1 System and Equipment Functions**

The purpose of the SWM is to control, collect, handle, process, package and temporarily store solid waste generated by the plant prior to shipping the waste offsite.

#### **11.4.1.1 Normal Functions (Non-Safety Category)**

The SWM processes the LWM filtering skid spent resins and filter backwash, lab waste, oily sump waste, filter backwash sludges, reverse osmosis concentrates, charcoal media, and bead resins generated by the FPC, ICC, and CFD systems. Contaminated solids such as HEPA and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also disposed of in the SWM system.

The SWM is capable of receiving, processing, and dewatering the solid radioactive waste inputs for permanent offsite disposal in accordance with CSA N292.0 (Reference 11.4-1).

#### **11.4.1.2 Normal Functions (Safety Category)**

The SWM has no Safety Category function. There is no liquid plant discharge from the SWM. Failure of the system does not compromise any SC system or component, nor does it prevent shutdown of the plant. No interface with the SC electrical system exists.

#### **11.4.1.3 Off-Normal Functions (Non-Safety Category)**

The SWM does not perform any non-safety category functions during off-normal conditions.

#### **11.4.1.4 Off-Normal Functions (Safety Category)**

The SWM does not perform any safety category functions during off-normal conditions.

### **11.4.2 Safety Design Bases**

The SWM system has no Safety Category function. There is no liquid plant discharge from the SWM system. Failure of the system does not compromise any SC system or component, nor does it prevent shutdown of the plant. No interface with the SC electrical system exists.

### **11.4.3 Description**

The SWM is primarily located in the RWB with individual collection locations around the plant, and consists of the following functions:

- Spent Resin Processing
- Sludge Processing
- Solid Waste Storage

Figure 11.4-1 provides a simplified single line diagram for the SWM. The subsystems, components and their functions are described in the following subsections.

#### **11.4.3.1 Equipment, Arrangement, and Location**

The SWM major components, which include the spent resin tank, sludge tanks, high integrity containers (HICs) (containers designed or engineered for radioactive waste), dewatering pump, sludge transfer pumps, and spent resin transfer pump are all located in the RWB which has adequate space for on-site storage of the HICs, shielded filter containers, and drums. The HICs are also located in such a manner as to allow for easy installation and removal.

#### **11.4.3.2 Major Components**

The following sections describe the major components in the SWM.

#### ***11.4.3.2.1 Spent Resin Processing***

##### **Spent Resin Tank**

The spent resin tank is a cylindrical stainless steel atmospheric tank fitted with a liquid level sensor and transmitter to prevent overfilling of the tank. The spent resin tank is also fitted with an overflow connection at least the size of the largest tank inlet and installed below the tank vent and above the high-level alarm setpoint. The tank overflow line is routed to the EFS system. A mixing eductor is fitted to the tank to facilitate mixing of the heavier solids and liquid and prevent heavier solids from settling at the bottom of the tank.

Tank capacities were selected to provide sufficient storage capacity for waste generation at normal generation rates and to allow sufficient time for the decay of shorter-lived radionuclides before shipping to an offsite facility.

##### **Spent Resin Tank Transfer Pump**

The SWM is designed with pumping capable of transferring the spent resin from the spent resin tank to the HICs.

#### ***11.4.3.2.2 Sludge Processing***

The following pieces of equipment are designed to exclusively handle sludge.

##### **Sludge Tank**

The sludge tanks are cylindrical stainless steel atmospheric tanks fitted with liquid level sensors and transmitters to prevent overfilling of the tanks. The sludge tanks are also fitted with an overflow connection at least the size of the largest tank inlet and installed below the tank vent and above the high-level alarm setpoint. The tanks' overflow lines are routed to the EFS system. Mixing eductors are fitted to the tanks to facilitate the creation of a slurry prior to the sludge transfer to the HICs.

Tank capacities were selected to provide sufficient storage capacity for waste generation at normal generation rates and to allow sufficient time for the decay of shorter-lived radionuclides before shipping to an offsite facility.

##### **Sludge Tank Transfer Pump**

The SWM is designed with pumps capable of transferring the sludge from the sludge tanks to the HICs.

#### ***11.4.3.2.3 Solid Waste Storage***

The following pieces of equipment are design to dewater the spent resin coming from the spent resin tank and the collected sludge from the sludge tanks.

##### **High Integrity Containers**

The SWM is designed with HICs configured with dewatering internals which remove standing liquid to <1% by volume. Per CSA N292.0 (Reference 11.4-1) solid radioactive waste should contain as little free standing and corrosive liquid as practicable. The HICs are installed with level sensors, level transmitters, passive vents, and sample monitoring.

##### **Dewatering Pump**

The SWM is designed with a pump capable of removing water from the HICs.

## **Drum Evaporator**

The drum evaporator is loaded with a drum filled with solids from lab samples, LWM skid waste, and oily waste from EFS. The drum evaporator is capable of removing excess moisture, through evaporation, which is routed to the RWB ventilation system to elimination.

### **11.4.3.2.4 Piping and Valves**

The piping and valves associated with resin and sludge systems are corrosion resistant stainless steel designed to the requirements of ASME B31.3 (Reference 11.4-2). The valves minimize crud buildup by utilizing full port valves for transferring, and diaphragm style valves to control flow. Full port valves reduce the potential for solidification due to minimal flow restrictions, and diaphragm valves can handle slurry fluids without the potential of contaminating working parts of the valve. Bottom valves on slurry containing tanks are designed to be flush mounted to eliminate pockets.

### **11.4.4 Materials**

Material and equipment selection for the system components is based on a planned 60-year design life, with appropriate provisions for maintenance and replacement. Piping and tanks are stainless steel for corrosion resistance.

Components which may require periodic replacement and/or maintenance prior to the end of plant life include, but are not limited to, electrical and electronics, O-rings, bearings, gaskets and seals, lubricants, valve disks, seats, and packing.

### **11.4.5 Interfaces with Other Equipment or Systems**

Refer to Table 11.4-1 for a listing and brief description of the SWM interfaces with other systems.

### **11.4.6 System and Equipment Operation**

The SWM operates continuously during all normal operating modes of the plant. The SWM carries out the following functions:

- Continuously remove dissolved and suspended solids, corrosion products, and spent bead resin so they do not enter the floor and equipment drain system
- Store high activity spent resin for decay
- Dewater and package the waste for temporary on-site storage prior to shipping the waste offsite

#### **11.4.6.1 Spent Resin and Sludge Processing**

Spent bead resin slurry from the ICC, CFD, FPC and LWM systems is sent to the dedicated spent resin tank for collection and decay. When sufficient bead resins have been collected in the spent resin tank, resin is transferred to HICs for dewatering and eventual disposal.

The associated tanks, piping, pumps, and valves can be flushed using CST water to minimize trace solids and reduce dose in the system.

Filter and demineralizer backwash from the FPC and CFD systems is collected in the two sludge tanks to allow the solids to settle out. Backwash water from the high flow fine filtration filters and skids can also be routed to the SWM sludge tanks for processing and liquid removal. Once enough sludge has been collected, the sludge is transferred by the sludge tank transfer pumps to the HICs for dewatering and eventual disposal.

The spent resin and sludge tanks are sized to accommodate the expected volume of waste generated in the upstream systems that feed waste into the SWM for processing.

#### **11.4.6.2 Solid Waste Storage**

Dry solid wastes consist of air filters, miscellaneous paper, rags, etc., from contaminated areas; containment clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory waste. These wastes are collected in containers located in appropriate areas throughout the plant, as dictated by volume of wastes generated during operation and maintenance. The filled containers are sealed and moved to controlled-access enclosed areas for temporary storage.

Most dry waste is expected to be sufficiently low in activity to permit temporary storage in an unshielded, cordoned-off area. Dry active waste is sorted and packaged in a suitably sized container that meets guidance in CSA 292.0 (Reference 11.4-1) for shipment to either a licensed offsite processor or for ultimate disposal. The dry waste is separated into three categories: non-contaminated wastes (clean), contaminated metal wastes, and the other wastes, i.e., clothing, plastics, HEPA filters, and other components. Non-contaminated (clean) materials identified during the sorting process can be removed for plant reuse or general debris disposal.

On-site storage space for packaged waste including the HICs, shielded filter containers, drums, and other shipping containers is provided in the RWB.

To the extent possible, discarded chemicals (including those classified as hazardous) will be kept out of the SWM. Waste is collected primarily in collection drums and sent offsite to an appropriately permitted vendor processor per CNSC REGDOC 2.11.1 (Reference 11.4-3).

Solids from lab samples, LWM skid waste, and oily waste from the EFS are collected in a drum and loaded in the drum evaporator. The drum evaporator removes excess moisture, through evaporation, which is routed to the RWB ventilation system for elimination.

#### **11.4.7 Instrumentation and Control**

##### **11.4.7.1 Instrumentation**

Level transmitters are provided on each tank for high-level indication and alarm to prevent overflow of the tanks and the HICs. As the HICs are intended to be removable, they come preinstalled with a level transmitter.

Other parameters indicated and alarmed in the MCR include pump pressures, discharge flows, remote valve positions, process mode, evaporator temperature and level.

##### **11.4.7.2 Control**

All operations, including system operating mode selection and transfer pump operation, can take place remotely through the MCR control panel. System interlocks prevent tank overflow and prevent simultaneous transfer of spent resin and sludge to the HICs. System logic includes a flushing cycle to clean the tanks and transfer lines of any residual particulates which are then transferred to the HICs for further dewatering. Flexibility in system operation exists to allow backflushing of filters into the empty sludge tank while the full sludge tank is being processed to the HICs. This flush consists of clean water addition from the CST. The addition of the flush water to the tanks assures complete mixing and forwarding of suspended solids minimizing the solids depositing within the transfer piping. Manual initiation is required to transfer the spent resin and sludge to the HICs.

#### **11.4.8 Monitoring, Inspection, Testing, and Maintenance**

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Isolation valves are provided for equipment, control valve, and piping circuit to facilitate maintenance and repair.

Tanks and pumps are arranged to provide adequate floor space and unobstructed clearance to permit monitoring, maintenance, and inspection of each unit.

The piping is routed to minimize the number of areas in the SWM requiring in-service inspection.

#### **11.4.9 Radiological Aspects**

SWM is designed to include shielding and controls sufficient to limit accessible general area radiation levels consistent with ALARA principle per CNSC REGDOC-2.7.1 (Reference 11.4-4) during normal processing. During filter media or waste container transfer operations, process system provisions for remote operation limit average worker radiation dose rates.

Isolation diaphragms are installed between all transmitters and the piping to avoid contaminating the instrument. Component selection is based upon the need for minimizing crud buildup.

There is no liquid plant discharge from the SWM. Failure of the system does not compromise any SC system or component, nor does it prevent shutdown of the plant. No interface with the SC electrical system exists.

#### **11.4.10 Performance and Safety Evaluation**

The SWM performs no Safety Category function.

#### **11.4.11 References**

- 11.4-1 CSA N292.0, "General principles for the management of radioactive waste and irradiated fuel," CSA Group.
- 11.4-2 ASME B31.3, "Process Piping," American Society of Mechanical Engineers.
- 11.4-3 CNSC Regulatory Document REGDOC-2.11.1, "Waste Management, Volume I: Management of Radioactive Waste."
- 11.4-4 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.4-1: SWM System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Non-Safety Class (SCN) Digital Control and Information System (DCIS)	NS DCIS provides necessary signals to run SWM	At SWM Equipment
Process Sampling (PS)	Process Sampling for radiological analysis, chemistry control, and Demineralizer performance monitoring	PS routed to RWB Sample Station
ICS Pool Cooling and Cleanup System (ICC)	ICC resin is sent to the spent resin tank	At SWM Equipment
Fuel Pool Cooling and Cleanup (FPC)	FPC resin is sent to the spent resin tank. Filter backwash water is sent to the sludge tanks	At SWM Equipment
Liquid Waste Management System (LWM)	Spent resin from LWM is sent to spent resin storage tank. Filter sludge and backwash sent to sludge tanks	At SWM Equipment
Condensate Filters and Demineralizers System (CFD)	CFD resin is sent to the spent resin tank. Backwash water is sent to the sludge tanks	At SWM Equipment
Plant Pneumatics System (PPS)	PPS supplies instrument air to the SWM AOV and dewatering pump	At SWM AOVs Dewatering Pump
Non-Safety Electrical Distribution System	Supplies power to SWM	At SWM Equipment
Heating, Ventilation, and Cooling System (HVS)	SWM vents to local HVS system	At SWM Equipment
Equipment and Floor Drain System (EFS)	Collects liquid waste from SWM	At EFS drain funnel



NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

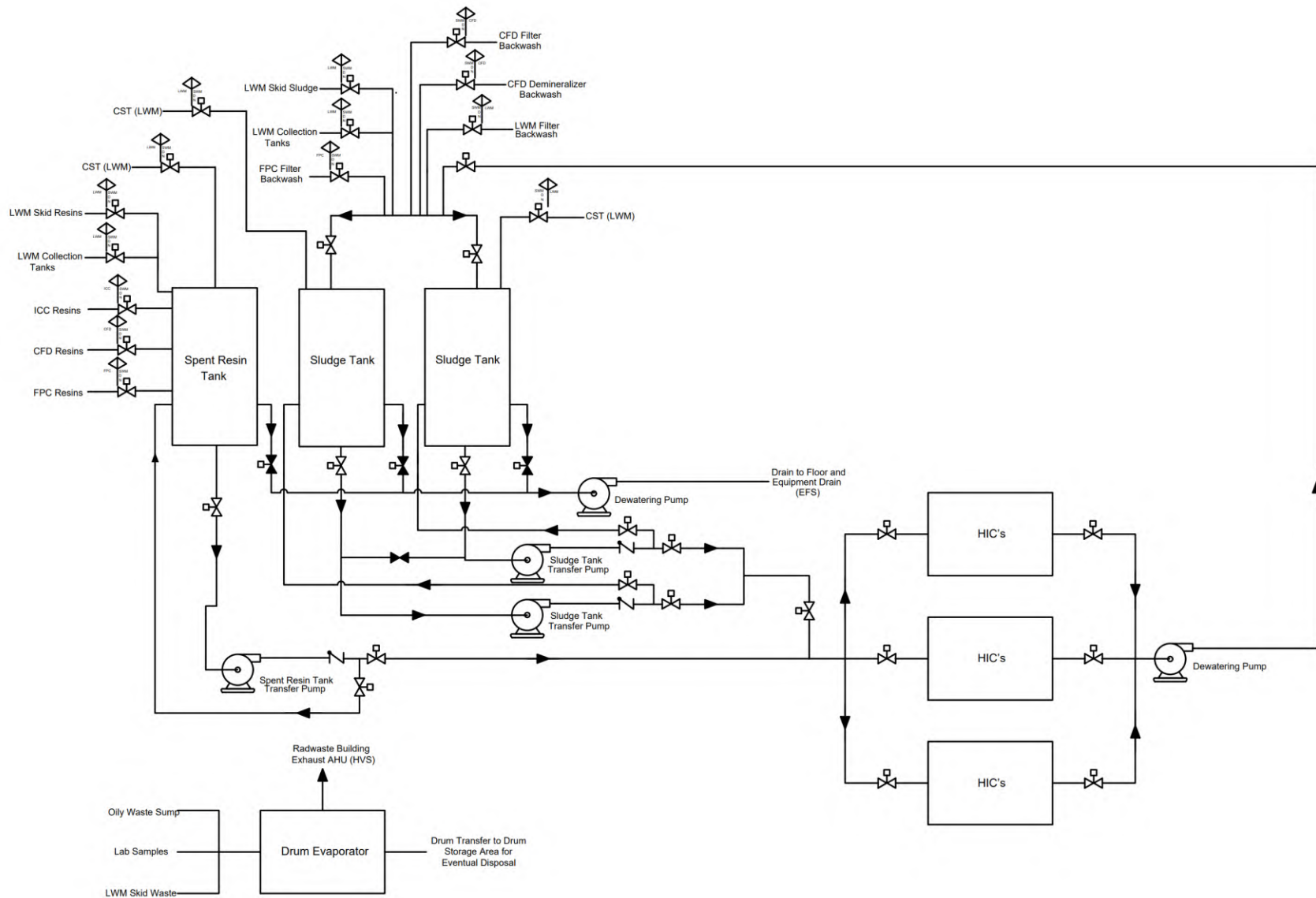


Figure 11.4-1: System Simplified Line Diagram

## **11.5 Process and Effluent Radiological Monitoring**

### **11.5.1 System and Equipment Functions**

The Process Radiation and Environmental Monitoring System (PREMS) for the BWRX-300 provides continuous and periodic monitoring to allow determination of the content of radioactive material in various gaseous, liquid process, and effluent streams.

PREMS also provides monitoring of other conditions such as oxygen and hydrogen concentrations in containment, HVAC contamination and toxic gas intake. PREMS includes a means to collect liquid and gaseous samples for analysis to support monitoring of plant processes.

PREMS consists of four subsystems as described below. The Process Radiation Monitoring subsystem is mostly relevant to process and effluent radiological monitoring with contributions from the other subsystems as discussed in detail in this Chapter. Detailed descriptions of the other subsystems of the PREMS are discussed in other PSAR chapters as indicated.

The PREMS includes the following subsystems:

- Process Radiation Monitoring Subsystem (PRM) – the purpose of the PRM is to determine the content of radioactive material in various gaseous and liquid process and effluent streams.
- Area Radiation Monitoring Subsystem (ARM) – the purpose of the ARM is to continuously measure, indicate and record gamma radiation levels at strategic locations throughout the plant, except for the containment areas, and initiates alarms in the MCR, Secondary Control Room (SCR), and locally to warn personnel. (Details of the ARM are found in Chapter 12, Subsection 12.3.14.)
- Containment Monitoring Subsystem (CMon) – the purpose of the CMon is to provide instrumentation to monitor multiple parameters inside containment. (Details of the CMon system are found in Chapter 12, Subsection 12.3.14.)
- Process Sampling Subsystem (PS) - The PS collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. (Details of the PS are found in Chapter 9A, Section 9A.3.1.).

#### **11.5.1.1 Normal Functions (Non-Safety Category)**

Radiation monitoring for plant operations provides personnel with sufficient warning to seek protection from radioactive material content in Non-Safety Category gaseous and liquid effluent, and process streams.

The following Non-Safety Category functions are monitored by this PRM instrumentation:

- Gaseous Effluent Streams:
  - Main Turbine Gland Seal Steam Condenser Exhaust
  - TB Normal HVAC Exhaust
  - TB Compartment Area HVAC Exhaust
  - TB Annex HVAC Exhaust
  - Radwaste Building HVAC Exhaust
- Gaseous Process Streams:
  - Offgas Pre-treatment

- Offgas Post-treatment
- Charcoal Vault Ventilation
- Liquid Process Streams:
  - CUW
  - Shut Down Cooling System

The following ARM area radiation monitors are provided:

- FPC Filter/Demineralizer
- Various locations in the Refueling Area
- Others based on plant layout

Other interfaces for the ARM include locations within:

- Reactor Building Structure
- Turbine Building Structure
- Control Building Structure
- Radwaste Building Structure

#### **11.5.1.2 Normal Functions (Safety Category)**

The following Safety Category functions are initiated by the PRM instrumentation:

- Main Steam Line (MSL) in the NBS downstream of the Containment Isolation Valves
- ICS leaking
- Liquid Radwaste Discharge
- CWS Discharge
- RB HVAC exhaust
- Stack
- Refuel Floor/Fuel Pool Area HVAC exhaust
- Containment Purge exhaust
- Control Room Habitability Area HVAC subsystem
- SCR HVAC

#### **11.5.1.3 Off-Normal Functions (Non-Safety Category)**

The off-normal non-safety category functions are generally the same as the normal non-safety category functions.

#### **11.5.1.4 Off-Normal Functions (Safety Category)**

The off-normal safety category functions initiated and monitored by the PRM instrumentation are the same as the normal safety category functions.

### **11.5.2 Safety Design Bases**

The PRM provides both Safety Category and Non-safety Category functions. The safety category functions include the environmental safety and radiation protection instrumentation functions to

initiate protective actions to limit the potential release of radioactive materials to the environment if major process/effluent stream radiation levels are exceeded. A second function is to provide plant personnel with alarms in support of personnel exposure prevention and potential environmental issues.

In accordance with CNSC REGDOC-2.5.2, Section 8.11 (Reference 11.5-1), liquid and gaseous radioactive release points/paths within the plant are identified and monitored by the PRM to ensure that planned radiological releases from both normal operation and waste management activities conform to the ALARA principle and are below prescribed limits. Action levels are established per CSA N288.8 (Reference 11.5-2) and monitored through the implementation of an Effluent Monitoring Program per CSA N288.5 (Reference 11.5-3). Details on the Effluent Monitoring Program are provided in Chapter 20, Section 20.11.2.

Radiation monitoring is provided during normal reactor operation, anticipated operational occurrences, and post-accident conditions.

### **11.5.3 Description**

The PRM provides radiation instrumentation required for:

- Environmental safety and protection
- Monitoring during plant operation

This instrumentation is provided in various plant locations for monitoring in various plant systems as described in Subsection 11.5.5.

### **11.5.4 Materials**

The materials selected for use in the system have been chosen to fulfill environmental requirements. Components and cables used in radioactive areas are qualified for the anticipated or calculated radiation conditions.

### **11.5.5 Interfaces with Other Equipment or Systems**

The PRM interfaces with other equipment and systems as a function of system operation. The interfaces are listed in Table 11.5-1 and summarized below.

#### **11.5.5.1 Nuclear Boiler System**

The PRM monitors the gamma radiation level of the steam transported by the Main Steam Lines (MSLs) in the MSL tunnel. The normal radiation level is produced primarily by coolant activation gases plus smaller quantities of fission gases being transported with the steam. The MSL radiation monitors consist of three instrument channels for each of the two main steam lines. Each channel consists of a local gamma-sensitive detector with instrumentation to transmit the data to the MCR and SCR.

The detectors are physically located near the MSLs just downstream of the outboard MSL isolation valves in the steam tunnel. These detectors are arranged so that they are capable of detecting significant increases in radiation levels in either of the MSLs in operation.

The PRM provides the detection signals to the DCIS that support decision logic for shutdown and isolation of the main turbine condenser mechanical vacuum pump and the SJAEs upon detection of high MSL radiation. Channel trips are annunciated in the MCR and SCR. The measurement range is selected to provide detection from normal background radiation at zero percent reactor power up to, and including, gross releases of fission products from reactor fuel into the reactor vessel and its subsequent transport to the MSLs.

#### **11.5.5.2 Isolation Condenser System**

The PRM provides radiation monitoring for the Isolation Condenser (IC) pool areas that are part of the ICC for monitoring the ICS system. The PRM will monitor the radiation from the exhaust of the air above each IC. Elevated radiation levels may indicate an ICS pressure boundary leakage. The radiation monitors will be placed above each inner pool, near the access hatch to facilitate easy access for maintenance. Any condenser leakage would release radionuclides into the atmosphere and the slightly positive pressure maintained inside the inner pool air space would force air through the ventilation space that will be monitored for radiation.

#### **11.5.5.3 Shutdown Cooling System**

The PRM provides a single gamma-sensitive radiation monitor located on the overboard line upstream of the SDC Supply Flow Element and provide alarm function only on a high radiation signal.

#### **11.5.5.4 Reactor Water Cleanup System**

The PRM provides a single gamma-sensitive radiation monitor that will be located between the CUW regenerative heat exchanger and the pressure reduction station and provide alarm function on a high radiation signal.

#### **11.5.5.5 Liquid Waste Management System**

The PRM provides a radiation monitor on the offsite discharge line from the LWM, with an automatic shutoff function that would terminate the discharge based on a determined setpoint in the radiation monitor.

#### **11.5.5.6 Offgas System**

The PRM provides radiation monitoring at various locations in the OGS.

##### ***11.5.5.6.1 Charcoal Vault Ventilation***

The ventilation of the Charcoal Vault is monitored for gamma radiation level, to detect charcoal tank leakage. A single instrument channel is used. The channel includes a gamma-sensitive detector and a radiation monitor. The detector is located outside the Charcoal Vault on the HVAC exhaust line from the vault. The radiation monitor is located adjacent to the ductwork.

##### ***11.5.5.6.2 Offgas Pre-treatment***

Radioactivity is monitored in the main turbine condenser offgas after it has passed through the offgas condenser and moisture separator/cooler. The single channel monitor detects the gross radiation level that is attributable to the fission gases that are produced in the reactor and then transported with steam through the turbine to the main turbine condenser.

A continuous sample is extracted from the offgas pipe, then passed through a sample chamber and a sample panel. Sample line flow is measured and indicated on the sample panel. A gamma-sensitive detector, positioned on the sample chamber, is connected to a local radiation monitor.

The radiation level reading can be directly correlated to the concentration of the noble gases in the sample chamber by obtaining a grab sample at the sample panel. The sample is then removed and analyzed with a multi-channel gamma pulse height analyzer to determine the concentration of the various noble gas radionuclides. A correlation between the observed activity and the monitor reading permits calibration of the monitor.

##### ***11.5.5.6.3 Offgas Post-treatment***

The OGS post-treatment stream is monitored for halogens, particulates, and noble gas releases during normal and accident conditions in the offgas piping downstream of the OGS charcoal

adsorbers and upstream of the OGS discharge valve. A continuous sample is extracted from the OGS piping, passed through an offgas post-treatment sampler for monitoring and sampling, and returned to the OGS piping. The sampler contains provisions for continuous gaseous, particulate and halogen radioactivity monitoring of the offgas post-treatment process. Automatic compensation for variation in main process flow is provided to maintain the sample panel flow proportional to the main flow. A local radiation monitor, connected to a gamma-beta sensitive radiation detector, analyzes, and visually displays the measured radiation level. The radiation monitor actuates MCR annunciators for operator response as discussed in Section 11.3.11.

#### **11.5.5.7 Main Turbine Equipment**

The PRM provides radiation monitoring for the Main Turbine Gland Seal Steam Condenser Exhaust flow path.

#### **11.5.5.8 Circulating Water System**

The PRM provides radiation monitoring CWS discharge to the NHS, with an automatic shutoff function that terminates the discharge based on a determined setpoint in the radiation monitor.

#### **11.5.5.9 Heating Ventilation and Cooling System**

##### ***11.5.5.9.1 Reactor Building Heating Ventilation and Cooling System***

The PRM and ARM provide monitoring in the fuel handling area of the RB. On a high radiation signal, RB supply and exhaust isolation dampers close. Supply and exhaust air handling units will de-energize.

The PRM provides three gamma-sensitive radiation monitors for each RB lower-level supply AHUs. Upon a high radiation signal, both RB lower-level supply AHUs de-energize and isolate, while the exhaust AHU will ramp down in flow. A high radiation signal causes an auto start of a SCR makeup AHU.

##### ***11.5.5.9.2 Control Building Heating Ventilation and Cooling***

The PRM provides an adequate number of gamma-sensitive radiation monitors commensurate with its SC for each Control Room Normal Supply AHUs. On a high radiation signal, both AHU inlet dampers close and de-energize, and an automatic swap over to the Control Room Envelope Emergency Filtration Units takes place.

The PRM provides an adequate number of gamma-sensitive radiation monitors, commensurate with its SC for each of the Control Room Envelope Emergency Filtration Units.

##### ***11.5.5.9.3 Turbine Building Heating Ventilation and Cooling***

The PRM and ARM provide general monitoring located in the TB. Upon receiving a high radiation signal from PRM, TB supply and exhaust isolation dampers close, and supply/exhaust AHUs de-energize. The planned monitoring is for noble gases, iodides, and particulates.

The TB Annex Exhaust AHU is also monitored. Upon a high radiation signal on the TB Annex Supply AHU, the respective inlet and exhaust dampers closes and the AHUs deenergizes.

The main Plant Vent Stack is located on the TB and is equipped with gamma-sensitive radiation monitor(s).

##### ***11.5.5.9.4 Radwaste Building Heating Ventilation and Cooling***

The PRM includes three radiation monitors on the combined AHU discharge duct and has no automatic actions on a high radiation signal.

#### **11.5.6 System and Equipment Operation**

During normal operations, the monitoring equipment is continuously in operation, or performing periodic monitoring as required, unless removed from service for maintenance. Sampling equipment operates continuously, periodically, or in the grab sample mode. There may be cases where the system with which the PRM interfaces is out of service, in which case the instrumentation allocated to that system will be taken out of service.

There are no generic startup protocols. Specific startup operation descriptions may be applicable to the host systems that interface with PREMS instrumentation.

The system performance requirements are generally the same for all operational modes.

#### **11.5.7 Instrumentation and Control**

PREMS instrumentation design is organized primarily by the Safety Category of the system functions that the instrumentation is supporting. PREMS provides monitoring only; there are no mitigating system functions controlled by PREMS. The PREMS may provide the primary detection in some cases or be a secondary avenue in other cases to collect instrumentation readings or signals from other systems for transmission to the DCIS.

##### **11.5.7.1 Instrumentation**

Each subsystem consists primarily of radiation detectors and transmitters. The sampling instrumentation primarily consists of grab samples which are analyzed to identify and quantify the specific radionuclides in effluents. The results from the sample analysis are used to establish relationships between the gross gamma monitor readings and concentrations or release rates of radionuclides in continuous effluent releases.

##### **11.5.7.2 Radiation Level Setpoints**

The derivation of trip setpoints for effluent and discharge radiation monitors are specified in plant operating procedures. Setpoints are established on the basis of discharge flow rates and the resulting offsite doses (as discussed in Chapter 20).

#### **11.5.8 Monitoring, Inspection, Testing, and Maintenance**

Maintenance includes servicing and replacement of defective components and adjustments as required. Periodic testing or calibration checks will be performed as part of the maintenance plan. If any work is performed that would affect the calibration of the instrument, a re-calibration is performed following the maintenance operation.

#### **11.5.9 Radiological Aspects**

The PREMS is comprised of subsystems that monitor liquid and gaseous effluents utilizing components designed and installed in various ways to minimize the possibility of contaminating non-radiological systems or components. To the extent possible, the system components and subsystems are constructed to avoid contamination due to leakage, spills, errors in valve lineup or other operating conditions as a result of interfacing with radioactive systems. These types of radiation monitoring components and subsystems are typically purely electrical in nature and do not physically interconnect with the radioactive systems they are monitoring. In addition, these PREMS subsystems do not interconnect with other non-radioactive systems, thereby eliminating the potential for transfer of radioactive material from a radioactive system to a non-radioactive system.

There are circumstances in the design of the PREMS subsystems where some interconnections to radioactive systems are necessary. In these cases, the additional subsystem interconnections to non-radioactive systems are limited to purge air, purge water and makeup water for filling loop

seals. The designs of these interconnections prevent contamination of the non-radioactive system or process due to leakage, spillage, valving errors, or other operating conditions.

PREMS is designed to minimize facility and environmental contamination, minimize the generation of radioactive waste, and facilitate decommissioning. These objectives are supported by the procedures for operation, and is controlled as follows:

- Minimizing contamination by:
  - Locating radiation detectors outside the process that they monitor, whenever feasible, to avoid the potential of coming in contact with a radioactive process
  - Providing atmospheric purging of the internal portion of air sampling skids as necessary
  - Providing the ability for liquid flushing of the internal portions of liquid sampling skids as necessary
  - Designing the interior portions of liquid and gaseous sampling chambers to minimize the plate out of radioactive material
  - Designing sample extraction points such that they minimize the potential for spillage and contamination of adjacent areas
- Facilitating decommissioning by providing equipment, where feasible, that reduces the need for decontamination during the removal and disposal of the equipment
- Minimizing the generation of radioactive waste by:
  - Directing continuous samples from radioactive processes back to the sampled process
  - Utilizing electronic devices that simulate radioactive sources, where compatible with the subsystem design, in order to minimize the use of radioactive sources
  - Minimizing the amount of a sample that needs to be extracted, consistent with laboratory and sensitivity requirements

#### **11.5.10 Performance and Safety Evaluation**

PREMS provides monitoring only; there are no mitigating system functions controlled by PREMS.

#### **11.5.11 References**

- 11.5-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 11.5-2 CSA N288.8, "Establishing and implementing action levels for releases to the environment from nuclear facilities," CSA Group.
- 11.5-3 CSA N288.5, "Effluent Monitoring Programs At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 11.5-4 Government of Canada SOR/2000-203, "Radiation Protection Regulations."



NEDO-33961 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 11.5-1: PRM Subsystem System Interfaces**

Interfacing System	Interface Description	Interface Boundary
Nuclear Boiler System (NBS)	Radiation monitoring of the MSLs (downstream of the CIVs)	MSL radiation detectors
Digital Control and Information Systems (DCIS)	DCIS SC1, SC2, and SC3, and SCN instrumentation communication	Instrumentation in PRM communicates with the appropriate DCIS system based on SC
ICS	Radiation monitoring of the ICS Pools	ICS Pool radiation detectors
Shut Down Cooling System (SDC)	Radiation monitoring of the discharge header	Discharge header radiation detector
Reactor Water Cleanup System (CUW)	Radiation monitoring between the regenerative heat exchanger and the pressure reduction station	Radiation detector between the regenerative heat exchanger and the pressure reduction station
Liquid Waste Management System (LWM)	Radiation monitoring of LWM discharge (supports automatic LWM isolation in the event of high radiation)	Discharge radiation detector
Offgas System (OGS)	Radiation monitoring of the Pre-Treatment, Post-Treatment, and Charcoal Ventilation streams	Pre-Treatment, Post-Treatment, and Charcoal Ventilation radiation detectors
Main Turbine Equipment	Radiation monitoring of the gland seal steam condenser exhaust	Gland seal steam condenser exhaust radiation detector
Circulating Water System (CWS)	Radiation monitoring of discharge to Normal Heat Sink (supports automatic CWS isolation in the event of high radiation)	Discharge radiation detectors
Heating Ventilation, and Cooling System (HVS)	Radiation monitoring of the RB, CB, TB and RWB HVAC systems; area radiation monitoring of the refuel floor/pool area (supports automatic HVAC isolation in the event of high radiation); toxic gas monitoring of the CB HVAC intakes	RB, CB, TB, and RWB HVAC radiation detectors; CB toxic gas monitors



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33962

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 12  
Radiation Protection**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
BWR	Boiling Water Reactor
CAM	Continuous Air Monitor
CB	Control Building
CFD	Condensate Filters and Demineralizers System
CIS	Containment Inerting System
CMon	Containment Monitoring Subsystem
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DNNP	Darlington New Nuclear Project
EFU	Emergency Filter Unit
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
GEH	GE Hitachi Nuclear Energy
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating Ventilation and Cooling System
HEPA	High Efficiency Particulate Air
ICRP	International Commission on Radiological Protection
ICS	Isolation Condenser System
ISI	In-Service Inspection
LOCA	Loss-of-Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
OGS	Offgas System
OLC	Operational Limits and Conditions
OPEX	Operating Experience

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
PAM	Post-Accident Monitoring
PLSA	Plant Services Area
POSAR	Pre-Operational Safety Analysis Report
PPE	Personal Protective Equipment
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
RB	Reactor Building
RHX	Regenerative Heat Exchanger
RPV	Reactor Pressure Vessel
RW	Radwaste
RWB	Radwaste Building
SC	Safety Class
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejector
TB	Turbine Building
URD	Utility Requirements Document
USNRC	U.S. Nuclear Regulatory Commission

## TABLE OF CONTENTS

<b>12.0 RADIATION PROTECTION .....</b>	<b>12-1</b>
12.1 Optimization of Protection and Safety .....	12-1
12.1.1 Design and Construction Policies .....	12-1
12.1.2 Operational Policies .....	12-1
12.1.3 Compliance With Standards for Radiation Protection to Ensure Occupational Exposures are ALARA .....	12-1
12.1.4 Ensuring Occupation Radiation Exposures Remain ALARA .....	12-2
12.1.5 ALARA Design Considerations .....	12-2
12.1.1 References.....	12-5
12.2 Sources of Radiation.....	12-6
12.2.1 Contained and Immobile Sources of Radioactive Material .....	12-6
12.2.2 Sources of Airborne Radioactive Material.....	12-10
12.2.3 Airborne Sources for Within the Plant.....	12-11
12.3 Design Features for Radiation Protection .....	12-13
12.3.1 Facility and Equipment Design Features .....	12-13
12.3.2 ALARA Design Guidelines for Mechanical System and Components ..	12-13
12.3.3 In-Service Maintenance and Inspection Provisions .....	12-20
12.3.4 Room Leak-Tightness and Drain Provisions.....	12-21
12.3.5 General Considerations of Systems that Process Radioactive Fluids ..	12-21
12.3.6 ALARA Structures Criteria .....	12-24
12.3.7 ALARA Criteria for Electrical Instrumentation and Control Design .....	12-28
12.3.8 System and Major Component Design Considerations .....	12-29
12.3.9 Radwaste Building .....	12-31
12.3.10 Reactor Building Spent Fuel and Components.....	12-32
12.3.11 Control Building.....	12-34
12.3.12 Decommissioning ALARA Considerations.....	12-34
12.3.13 Ventilation .....	12-35
12.3.14 Monitoring of Individuals and Working Areas (Instrumentation for Radiation Levels and Airborne Radioactivity) .....	12-36
12.3.15 References.....	12-39
12.4 Shielding .....	12-40
12.4.1 General Design Guides.....	12-40
12.4.2 Design Description .....	12-40

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

12.4.3	Shielding Method Design .....	12-41
12.4.4	References.....	12-41
12.5	Dose Constraints and Dose Assessment.....	12-42
12.5.1	Interfaces for Protecting People and the Environment.....	12-42
12.5.2	References.....	12-44
12.6	Radiation Zones and Access Requirements .....	12-45
12.6.1	Radiation Zones .....	12-45
12.7	Radiation Protection Program .....	12-66
12.7.1	References.....	12-66



**LIST OF TABLES**

Table 12.6-1: BWRX-300 Radiation Zone Classifications .....	12-47
---	-------

## LIST OF FIGURES

Figure 12.6-1: Reactor Building Level -34.0 Meters Radiation Zones .....	12-48
Figure 12.6-2: Reactor Building Level -29.0 Meters Radiation Zones .....	12-49
Figure 12.6-3: Reactor Building Level -21.0 Meters Radiation Zones .....	12-50
Figure 12.6-4: Reactor Building Level -14.5 Meters Radiation Zones .....	12-51
Figure 12.6-5: Reactor Building Level -8.5 Meters Radiation Zones .....	12-52
Figure 12.6-6: Fuel Pool Mezzanine - -4.8 Meters Radiation Zone .....	12-53
Figure 12.6-7: Reactor Building Level 0.0 Meters Radiation Zones .....	12-54
Figure 12.6-8: Reactor Building Level 4.9 Meters Radiation Zones .....	12-55
Figure 12.6-9: Reactor Building Level 13.0 Meters Radiation Zones .....	12-56
Figure 12.6-10: Reactor Building Level 20.8 Meters Radiation Zones .....	12-57
Figure 12.6-11: Turbine Building Level 0.0 Meters Radiation Zone.....	12-58
Figure 12.6-12: Turbine Building Level 6.1 Meters Radiation Zone.....	12-59
Figure 12.6-13: Turbine Building Level 12.2 Meters Radiation Zone.....	12-60
Figure 12.6-14: Plant Services Building 0.0 Meters Radiation Zone .....	12-61
Figure 12.6-15: Radwaste Building Level 0.0 Meters Radiation Zone .....	12-62
Figure 12.6-16: Radwaste Building Level 6.1 Meters Radiation Zone .....	12-63
Figure 12.6-17: Radwaste Building Level 13.0 Meters Radiation Zone .....	12-64
Figure 12.6-18: Control Building Level 0.0 Meters Radiation Zone .....	12-65

## **12.0 RADIATION PROTECTION**

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel is kept As Low As Reasonably Achievable (ALARA). The systematic application of the ALARA principle during the design phase of the BWRX-300 establishes the basic design criteria observed to reducing occupational exposure during plant operation and maintenance, decommissioning and post-accident ALARA.

ALARA design requirements are established to improve the layout of enclosures, accesses, and exits from controlled areas of the plant structures that confine radioactive material. The design of plant Structures, Systems, and Components (SSCs) minimizes personnel exposure to radiation during operation, inspection, maintenance, or plant design modifications.

The ALARA design requirements keep radiation exposures ALARA during normal operation or Anticipated Operational Occurrences (AOOs) and planned radioactive material releases below regulatory limits. The ALARA design criteria includes provisions for mitigating the radiological consequences of design basis accidents in accordance with Canadian Nuclear Safety Commission (CNSC) REGDOC-2.5.2, Section 4.1.1 (Reference 12.1-1).

The plant design:

- Precludes the release of radioactive material to the public and the environment that exceeds the limits of applicable regulations for normal operations, transients, and accidents
- Minimizes personnel exposure
- Minimizes the generation of radioactive contamination and waste

### **12.1 Optimization of Protection and Safety**

#### **12.1.1 Design and Construction Policies**

ALARA is applied during the initial design of the plant and implemented via internal design reviews. The design is reviewed in detail for ALARA considerations and is reviewed, updated, and modified during the design phase to apply best practices operating experience. Engineers review the plant design and integrate the layout, shielding, ventilation and instrument monitoring designs with traffic control, security, access control, and health physics aspects to ensure the overall design maintains exposures ALARA.

All radioactive fluid pipe expected to contain significant radiation sources are designed with adequate shielding and properly routed to reduce personnel exposure. Operating Experiences (OPEX) from industry that includes the operating Boiling Water Reactor (BWR) fleet are continuously integrated during the design phase.

#### **12.1.2 Operational Policies**

The ALARA principle is applied during the operation of the BWRX-300 throughout the completion of the plant life cycle by implementing the Radiation Protection Program discussed in Section 12.7.

#### **12.1.3 Compliance With Standards for Radiation Protection to Ensure Occupational Exposures are ALARA**

The BWRX-300 plant is designed for Radiation Protection to comply with Government of Canada Radiation Protection Regulations (Reference 12.1-2) and to meet the requirements and guidance in CNSC REGDOC-2.7.1 (Reference 12.1-3). Additionally, guidance from International Atomic

Energy Agency (IAEA) international safety standards, and U.S. Nuclear Regulatory Commission (USNRC) guidance informs the design.

#### **12.1.4 Ensuring Occupation Radiation Exposures Remain ALARA**

The BWRX-300 design considers that the majority of station personnel radiation exposure of is received during:

- Maintenance
- Radwaste (RW) work
- In-service inspection
- Refueling
- Non-routine operations
- Decommissioning

The BWRX-300 SSC that contain, collect, store, process, or transport radioactive materials follow the ALARA principle.

The BWRX-300 is designed to meet requirements in REGDOC 2.5.2, Section 8.13 (Reference 12.1-1). REGDOC-2.7.1, Section 4.1 (Reference 12.1-3) provides measures for the application of ALARA with reference to international standards for facility, equipment, and instrumentation design features delineated in Section 12.3. Cited standards for implementing and for maintaining ALARA include IAEA NS-G-1.13 (Reference 12.1-4) and International Commission on Radiological Protection (ICRP) standards such as ICRP Publication 101(b), (Reference 12.1-5) and ICRP Publication 103, (Reference 12.1-6).

The Radiation Protection Program discussed in Section 12.7 is implemented in accordance with REGDOC 2.7.1 (Reference 12.1-3) to ensure that radiation exposures to personnel and the public are ALARA, and that personnel are qualified, trained and monitored to keep exposures ALARA.

#### **12.1.5 ALARA Design Considerations**

This section describes the methods and features incorporated into the design that keep exposures ALARA.

##### **12.1.5.1 General Design Considerations for ALARA Exposures**

Consistent with the recommendations of IAEA and ICRP standards, general design considerations and methods employed to maintain in-plant radiation exposures ALARA, have two objectives:

- Reducing the necessity for and amount of personnel time spent in radiation areas
- Reducing radiation levels in routinely occupied plant areas in the vicinity of plant equipment that require personnel attention

Equipment and facility designs are evaluated for maintaining exposures ALARA during plant operations. Plant operations include normal maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, and decommissioning.

Descriptions of general design features that maintain doses ALARA during normal power and shutdown operations are provided in Section 12.3.

Features that assist in maintaining low occupational exposures during decommissioning include:

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

- Provisions for draining, flushing, and decontaminating equipment and piping
- Equipment design minimizes the radioactive material buildup and facilitates crud traps flushing
- Shielding protection during maintenance, repairs, or during decommissioning operations
- Provision for adequate space for movable shielding
- Separation of highly radioactive equipment from less radioactive equipment
- Provision for separate shielded compartments for adjacent items of radioactive equipment
- Provision for access hatches for the installation or removal of plant components
- Provision for the Reactor Water Cleanup (CUW) and Shutdown Cooling (SDC) System and the condensate demineralizer to minimize crud buildup

#### **12.1.5.2 BWRX-300 Design Considerations**

The following BWRX-300 design features minimize radioactive contamination:

1. Containment in areas where leaks and spills are likely to occur.
2. Leak detection capability to provide prompt detection SSC leakage.
3. Usage of leak detection methods (e.g., instrumentation, automated samplers) capable of early leak detection in areas difficult (inaccessible) to conduct regular inspections (such as the fuel pool), and buried, embedded, or subterranean piping) to avoid release of contamination. All BWRX-300 tanks containing radioactive fluids are within the Radwaste Building (RWB) that have cubicles and drain back into the radioactive liquid waste for processing.
4. Minimizing embedded piping, sumps, or buried equipment to facilitate decommissioning.
5. Removal or replacement of equipment or components during facility operation or decommissioning.
6. Minimizes the generation of radioactive contamination and waste during operation and decommissioning by reducing the volume of components and structures that become contaminated during plant operation.

#### **12.1.5.3 ALARA Equipment Design Considerations**

BWRX-300 engineering design procedures require the integration of international standards (including USNRC Regulatory Guide 8.8), and the radiation received for the component or system application.

##### ***12.1.5.3.1 Equipment Design Consideration to Limit Time Spent in Radiation Areas***

BWR fleet OPEX is factored into current designs. Equipment instrumentation and controls are accessible during normal and abnormal operating conditions. Equipment such as the CUW and the Fuel Pool Cooling and Cleanup System (FPC) are remotely operated, including backwashing and precoat operations.

Equipment design facilitates maintenance. Equipment such as the Isolation Condenser System (ICS) heat exchanger is designed with an excess of tubes in order to permit tube plugging, when necessary. Heat exchanger drains exist on the shell side. Some valves have stem cartridge-type packing that is easily replaced. Refueling tools come equipped with drainage and have smooth surfaces that reduces contamination. Vessel and piping insulation is easily removed.

The materials selected are chosen for operating conditions. Valves use stem packing to reduce leakage and maintenance.

#### **Equipment Design Consideration to Limit Component Radiation Levels**

Equipment and piping reduce the accumulation of radioactive materials in the equipment. The piping, where possible, is constructed of seamless pipe as a means to reduce radioactivity accumulation on seams. The filter demineralizers in the CUW and FPC systems are backwashed and flushed prior to maintenance.

Equipment is designed for limiting leaks or controlling fluid leaks include piping the released fluid to the sumps and using drip pans with drains piped to the floor drains.

The primary coolant system materials selected consist mainly of austenitic stainless steel, carbon steel and low alloy steel components.

The system design includes a CUW system on the reactor coolant. This system is designed to limit the radioactive isotopes in the coolant by processing the coolant in the plant Condensate Filters and Demineralizers System (CFD) for cleanup. The feedwater piping downstream of the CFD is stainless steel with low cobalt content. Zinc is injected into the system to aid in reducing dose rates from Co-60 deposition.

#### **12.1.5.4 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA**

Facility general design considerations minimizing personnel time spent in radiation areas include:

- Locating equipment, instruments, and sampling stations, that require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas
- Laying out plant areas allowing remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Providing, where practicable, transportation of equipment or components requiring service to a lower radiation area
- Minimizing radiation levels in plant access areas and vicinity of equipment

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include:

- Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high radioactive fluids not passing through occupied areas)
- Providing adequate shielding between radiation sources and access and service areas
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone
- Providing central control panels permitting remote operation of Safety Class (SC) instrumentation and controls from the lowest radiation zone practicable
- Separating highly radioactive equipment from less radioactive equipment, instruments, and controls
- Providing adequate space for movable shielding for sources within the service area
- Providing and facilitating contamination control

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

- Providing service area decontamination
- Providing space for pumps and valves outside highly radioactive areas for maintenance and inspection
- Providing remotely operated centrifugal discharge and/or back-flushable filter systems for highly radioactive RW and cleanup systems
- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms
- Providing adequate space in labyrinth entrances for easy access
- Maintaining ventilation airflow patterns from areas of lower radioactivity to areas of higher radioactivity

**12.1.1 References**

- 12.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 12.1-2 Government of Canada,, SOR/2000-203, Radiation Protection Regulations.
- 12.1-3 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.1-4 IAEA Safety Standards Series No. NS-G-1.13, "Radiation Protection Aspects of Design for Nuclear Power Plants," International Atomic Energy Agency.
- 12.1-5 ICRP Publication 101(b), "The Optimization of Radiological Protection – Broadening the Process," International Commission on Radiological Protection.
- 12.1-6 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.

## **12.2 Sources of Radiation**

This section contains a qualitative description of radiation sources present in the BWRX-300 plant. These sources are identified based upon GEH BWR design evolution and fleet operations.

At this stage of the plant design, source strengths have not been calculated. The characterization of plant sources is limited to source descriptions. These sources are quantified in the Pre-Operational Safety Analysis Report (POSAR).

### **12.2.1 Contained and Immobile Sources of Radioactive Material**

#### **12.2.1.1 Containment Radiation Sources**

This section describes the significant radiation sources found inside the BWRX-300 containment. The calculated sources provided in the POSAR consist of those elements that contain significant quantities of radioactive elements. The radiation source does not include incidental contamination, such as sources in valves (deposition from corrosion), fission product species on component surfaces, or gamma sources emitted from pipes containing radioactive fluids. The Fine Motion Control Rod Drive (FMCRD) system provides the only other notable source of radiation in containment. The only significant source of post-operation radiation in containment is the reactor core and irradiated reactor internals.

#### **Reactor Vessel Core Sources**

A reactor vessel model and pertinent data necessary to calculate neutron and gamma fluxes inside and outside of the reactor core during normal operation is defined. Ex-core particle fluxes from the reactor core during operation calculations require a detailed analysis of neutron particle transport and using a deterministic solution to the Boltzmann equation or probabilistic modeling techniques. The primary source for the neutron and gamma fluxes outside the core is the fission process. Gammas are also created by the decay of fission products, and secondary gammas resulting from neutron absorption and scattering in structural materials, both inside and outside of the core. Nuclide cross-section libraries contain gamma production data for all of these sources. As a result, it is only necessary to define the neutron fission source in the core, and then perform a coupled neutron-gamma transport calculation.

After shutdown, the neutron fluxes are negligible, and N-16 activity quickly decays to zero. The most significant radiation source is the gammas resulting from fission product decay in the reactor core. Neutron and gamma fluxes inside and outside the reactor core during normal operation are provided in the POSAR that include neutron and gamma flux results, reactor vessel model definition, and all pertinent data used in the analysis for the reactor during operation and shutdown conditions.

#### *Physical Data*

A POSAR table presents the physical data required to form a model that provides sufficient regions to adequately represent the reactor. The in-core region is divided into a determined number of axial nodes, with one radial node per fuel bundle. A unique neutron fission source is determined for each of these nodes using the nodal cycle average power and exposure data. Water densities are determined at each of the determined planes for peripheral bundles and in-core bundles. A POSAR table provides nominal dimensions and material volume fractions for each boundary and region in the reactor model with core average data presented for the core. To describe the reactor core, thermal power, power density, core dimensions, core average material volume fractions, and cycle average reactor power distributions and exposures are provided. Reactor power distributions are given for both radial and axial distributions and represent cycle averages for an equilibrium cycle.



### *Core Boundary and Vessel Neutron Fluxes*

A POSAR table presents multigroup neutron fluxes at the representative location of the core boundary and at the vessel. The multigroup neutron fluxes and the fast neutron flux ( $E > 1$  MeV) at the peak elevation of the core boundary, vessel inside surface, and a determined thickness of the vessel are presented the POSAR. The uncertainty of the fast neutron flux at the vessel is estimated within a determined percentage. Normalized axial variations for the fast flux at the vessel inside surface are shown in another POSAR table.

### *Gamma Ray Source Energy Spectra*

The average gamma ray source energy spectra in both core and non-core regions is presented in a POSAR table. The energy spectrum in the core, bypass water, shroud, downcomer, and Reactor Pressure Vessel (RPV) is presented. This represents the average gamma ray energy released by energy group per unit volume of the region. The energy spectra in MeV per seconds per  $\text{cm}^3$  is used with the power distributions to obtain the source in any part of the core.

The gamma ray energy spectrum includes the fission gamma rays, the fission product gamma rays, and the gamma rays resulting from inelastic neutron scattering and neutron capture. The total gamma ray energy released in the core is estimated accurate within a conservative percentage.

### *Post-Operation Gamma Sources*

A gamma ray energy spectrum in MeV/sec per MW thermal in spent fuel as a function of time after operation is provided in a POSAR table. The data is prepared from the irradiation and decay calculation of a representative BWRX-300 fuel bundle average exposure. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel bundle is obtained by using the gamma ray energy spectra and the thermal power of the element during operation. Similar information suitable for the spent fuel pool fuel bundles with an average exposure is also provided in the POSAR.

### *Neutron and Gamma Ray Fluxes Outside the Vessel*

The maximum axial neutron and gamma ray fluxes outside the vessel is provided in the POSAR. The maximum axial flux occurs near the core midplane elevation where the maximum power density is located for the peripheral bundle. This elevation is determined using the data for the maximum axial neutron and gamma ray fluxes outside the vessel. The fluxes at this elevation represent the fluxes at the peak azimuth angle. The gamma ray calculations include gamma ray sources from regions inside the vessel and the vessel itself.

### **Control Rod Drive System Sources**

The Control Rod Drive (CRD) source term data is provided in the POSAR.

Historically, removal and replacement of CRDs is a significant contributor to occupational exposures. Improved maintainability associated with the FMCRDs design reduces personnel exposures compared to conventional BWR CRDs. The CRD system is comprised of three major elements: (a) the FMCRD mechanisms; (b) the Hydraulic Control Unit assemblies; and (c) the CRD hydraulic subsystem. The BWRX-300 CRD hydraulic subsystem supplies high-pressure demineralized water that is regulated and distributed to provide charging of the Hydraulic Control Unit scram accumulators and purge water flow to the FMCRDs. The CRD system is described in Section 4.5.

### **Reactor Startup Source**

The Cf-252 reactor startup source is shipped to the site in a special shielded cask. The source is transferred under water while in the cask and loaded into a stainless steel source holder. This is loaded into the reactor while remaining under water. The source and source holder are removed from the reactor during the first refueling outage and moved to a designated location in the fuel pool. Operations and Radiation Protection personnel determine placement and duration of residence for the Cf-252 source and holder in the fuel pool. The activity requirements for the reactor startup source are provided in the POSAR

#### **12.2.1.2 Reactor Building and Plant Services Area Radiation Sources**

The Reactor Building (RB) radiation source terms consist of significant quantities of radioactive elements from the fission source, but do not include sources due to incidental contamination such as sources in valves and pipes due to deposition of corrosion or fission products species on the surfaces of the components. Water radioactive source concentrations in the fuel pool are assumed as 1% of normal reactor water concentrations. The Plant Services Area includes the RB auxiliary bay where new and spent fuel enter and leave the RB. Source terms in the RB are identified in the POSAR.

### **Fuel Pool Cooling and Cleanup System**

The FPC system is designed to service the fuel pool, cask pit, and reactor cavity pool on a rotating basis. The activity in this system is the result of the accumulation of residual activity in each of these pools. The filters are backwashed and the backwash routed to the liquid RW systems for processing. Clean water connections are provided for this system to flush lines prior to switching between pools to prevent ancillary contamination between pools.

### **Fuel Pool**

The gamma source energy spectra from spent fuel in the fuel pool is significant; however, the spent fuel is submerged in water and shielding around the fuel pool is provided ALARA. Radiation source terms from the fuel pool is provided in the POSAR.

### **Reactor Building and Containment Post-Accident Radiation Source**

The BWRX-300 design limits potential radiation exposure from accidents to personnel and to the public in the plant using passive safety features, containment, and treatment of potential normal operation and accident sources. The following describes those features of the BWRX-300 germane to post-accident radiation sources in containment and the RB.

1. Containment is a nitrogen-inerted, Steel-Plate Composite Containment Vessel (SCCV) pressure boundary capable of containing all design basis accident sources with minimal leakage to the environment or other plant areas. The containment is provided with redundant passive cooling systems to ensure that this primary boundary does not exceed design criteria. Radioactive sources from the pressure vessel are adequately contained during design basis accidents and some beyond design basis accidents.
2. Surrounding the cylindrical containment, the BWRX-300 employs a RB that provides a secondary holdup volume to trap containment penetration and valve leakage. All major connections from containment (e.g., main steam, feedwater) include an isolation valve in the RB. Pipe ruptures within containment are mitigated by the RPV isolation valves for large and intermediate pipe ruptures, and the coolant released that becomes airborne is captured in the containment airspace. Any releases from potential ruptures outside of containment but inside the RB are isolated by the RPV isolation valves.

### **12.2.1.3 Turbine Building Source Terms**

Significant radioactive source terms in the BWRX-300 Turbine Building (TB) consist of those elements that contain significant quantities of radioactive materials. This does not include sources due to incidental contamination such as valves with corrosion deposition or fission products species on component surfaces. The following systems that may contain radioactive material sources are located in the TB.

#### **Reactor Water Cleanup System (CUW) Radiation Sources**

The CUW system is located in the TB in a shielded area containing other radioactive equipment and systems. Radioactive sources contained in the CUW result from component contamination by reactor water through the system and accumulation of radioisotopes removed from the water. Components for this system include a Regenerative Heat Exchanger (RHX), pressure reduction station, and valves. The RHX is fitted with decontamination ports. The CUW system does not have filters and demineralizer beds; however, the CFD provides all of the cleanup of the reactor coolant.

#### **Heating, Ventilation and Cooling System**

The Heating Ventilation and Cooling System (HVS) employs High Efficiency Particulate Air (HEPA) filters for use in the event of airborne contamination of the RB or controlled purge of containment. The HEPA filters, which can accumulate radioactive sources, are shielded to ensure the radiation zones are maintained ALARA in areas adjacent to the filter housings.

#### **Main Steam and Feedwater Lines**

All radioactive material in the main steam system results from radioactive sources carried over from the reactor core during plant operations. In most components carrying live steam, N-16 is the dominant source of radioactivity. Otherwise, under conditions where sufficient decay time has removed the N-16 source, noble radioactive gases become the dominant source term.

#### **Normal Operating Sources**

N-16 in the steam flow from the pressure vessel is the primary TB source of radioactivity. The N-16 source results in significant gamma shine from the main steam lines and steam bearing components. Other sources of radiation in the TB are the Offgas System (OGS) and CUW. The OGS consists of the recombiner, OGS condenser, and offgas charcoal tanks (located in the RWB), and the CUW includes an RHX, pressure reduction station, and various valves.

#### **Post-Accident Sources**

The TB contains no major post-accident sources of releasable radioactivity (discounting N-16 because of the short half-life) and potential releases are limited to liquid releases of low activity water. Two other sources exist that contain radioactive species, but in a form not amenable for release. The potential for accident releases from these two sources, the OGS, and the CUW system, is reduced due to heavy shielding and compartmentalizing of the components.

### **12.2.1.4 Radwaste Building Sources**

This subsection provides brief descriptions of the significant radioactive source terms found in the BWRX-300 RWB. These source terms consist of those elements that contain significant quantities of radioactive materials but do not include sources due to incidental contamination, such as sources in valve corrosion deposition or fission product species on the surfaces of the components.

## **Normal Operating Sources**

Major RWB components that contain radiation sources include liquid RW collection tanks, spent resin tanks, sludge tanks, liquid waste sample tanks, filtration equipment, and high integrity containers with solid waste. These contained sources will be quantified as the design progresses. These sources are based on the waste stream concentrations fed to the RW systems and represent sources for shielding calculations.

## **Post-Accident Sources**

Potential releases in the RWB are contained by isolating the RWB atmosphere and sealing any water releases in the building. The RWB is seismically designed, and the concrete tank areas are provided with sealant and steel liners to prevent any releases from high-activity areas.

### **12.2.2 Sources of Airborne Radioactive Material**

#### **12.2.2.1 Reactor Building**

The RB HVS is discussed in Subsection 9A.5.1. Subsection 12.2.1.3 discusses the radiation control aspects of the HVS.

#### **Normal Operation**

The main source of airborne activity in the RB is primary coolant leakage. The highest potential activity sources are in containment but pose no threat to plant personnel during normal operation.

The containment cooling system conditions and circulates air through the contaminated areas of the building. Flow is directed from the corridors (point of highest pressure) to the equipment alcove rooms, the rooms themselves, and finally to the external wall pipe chases and from the pipe chases back to the HVS.

Access into the containment is not permitted during normal operation. The ventilation system inside merely circulates the air, without filtering it. The only airflow out of containment into accessible areas is minor potential leakage through penetrations. During maintenance, the containment atmosphere is purged using the containment inerting system and replaced with clean air before access is permitted.

#### **Reactor Building Airborne Sources During Refueling**

BWR operating experience shows that airborne radioactivity is generated from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources occur during vessel head venting and fuel movement. The airborne radioactive material sources resulting from reactor vessel head removal are minimized by venting to the containment cooling system or to the main condenser prior to removal. The contribution to the airborne radioactivity due to the reactor vessel internals is minimized because they are wet and submerged.

Airborne radioactivity during refueling is similar or lower due to improved design features and OPEX incorporation. Experience shows that airborne radioactivity results from water in the reactor cavity exceeding 38°C (100°F) and flaking of cobalt dioxide (CoO<sub>2</sub>) from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources resulting from reactor vessel head and internals removal are determined from OPEX. OPEX shows that Iodine-131, Co-60, Mn-54, Nb-95, Zr-95, Ru-103, and Ce-144 are the major radioisotopes found with Ce-141, Cs-137, Co-58, and Cr-51 at lower concentrations.

To minimize airborne radioactivity, the following actions are specified:

- Maintain steam dryer and separator surfaces wet or covered
- Cool fuel pool through large heat capacity heat exchangers

- Fuel pool ventilation system designed to sweep the pool surface and prevent pool releases from mixing with the area atmosphere

### **Refueling Area of Reactor Building**

The refueling area is serviced by the containment cooling system, including radiation control aspects of the system is discussed in Chapter 9A, Subsection 9A.5.6.

The airborne activity source in the fuel pool area of the RB is in the spent fuel storage pool and equipment areas. The ventilation system is designed to sweep air from the fuel pool surface removing the major portion of potential airborne contamination. In addition, evaporation from the fuel pool is minimized by pool cooling.

The assumptions and parameters used to determine the airborne activity levels in the spent fuel storage pool and equipment areas, and the airborne concentrations are provided in the POSAR.

#### **12.2.2.2 Turbine Building Airborne Sources**

The main potential source of airborne radioactivity within the TB is leakage from valves on large steam pipes. The design provides for collection of this leakage and its transport back to the condenser. Therefore, noble gas airborne concentrations are negligible throughout the TB except for inside the Steam Jet Air Ejector (SJAE) cubicles. These areas are not normally occupied during operation, and the exhaust from these cubicles is exhausted to the environment after filtration eliminating the possibility of contamination of adjoining areas. Another airborne activity source is equipment leakage in the TB atmosphere.

#### **12.2.2.3 Radwaste Building Airborne Sources**

The RWB HVS is discussed in Chapter 9A, Subsection 9A.5.3.

RWB corridors and routine access operating areas do not have significant airborne radioactivity levels. Equipment cubicles are infrequently accessed and may contain low levels of airborne radioactivity, but design provisions minimize the release of radioactivity.

RWB tanks are filled from the top and as the water splashes into the tanks, dissolved and entrained radioactivity may become airborne. This activity is not released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. Pumps and valves for radioactive systems in the RWB are in separate compartments that are not normally accessed by operating personnel. The RWB ventilation design provides airflow from areas of low potential for airborne contamination to areas of increasing potential. This ensures that any leakage from RW pumps and valves is not directed into normally occupied areas of the building but exhausted from the building.

#### **12.2.2.4 Control Building Airborne Sources**

The Control Building (CB) HVS is discussed in Chapter 9, Subsection 9A.5.2. If the CB air intake radiation monitor setpoint is reached, the intake is isolated, and the system initiates the recirculation mode. The SCR is available if the MCR becomes uninhabitable. There are no contaminated areas in the CB.

#### **12.2.2.5 Plant Services Area**

The Plant Services Area (PLSA) contains the Reactor Auxiliary Bay where new and irradiated fuel is moved in and out of the reactor building.

### **12.2.3 Airborne Sources for Within the Plant**

The design keeps all radioactive material contained. Leaks from process systems, refueling, and decontamination may lead to airborne radioactivity. Equipment cubicles, corridors, and areas

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

routinely occupied by operating personnel do not contain significant airborne radioactivity sources and comply with the limits specified in SOR/2000-203 (Reference 12.2-2). Radioactive equipment that potentially leaks is installed in separate shielded compartments that are not routinely occupied.

In general, airflow within the building ventilation systems is from areas of low potential for airborne contamination to areas of increasing potential. Routinely occupied areas are maintained at low airborne radioactivity levels. OPEX data from operating BWRs corroborate the general lack of airborne activity in corridors and routinely occupied operating areas. Air samples and surface contamination swipe samples are performed to verify the absence of airborne and surface contamination.

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases remain airborne and are released to the atmosphere with little delay via ventilation exhaust ducts. Other radionuclides partition between air and water may plate-out on metal surfaces, concrete, and paint. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine (particulate, elemental, and hypoiodous acid forms).

### **12.3 Design Features for Radiation Protection**

The design features incorporated in the BWRX-300 are based on GEH BWR OPEX. The BWRX-300 ALARA design requirements incorporate the guidance of REGDOC-2.7.1 Radiation Protection (Reference 12.3-1).

#### **12.3.1 Facility and Equipment Design Features**

The BWRX-300 plant design features ensure plant personnel radiation exposures are ALARA. General design considerations and methods to maintain in-plant radiation exposures ALARA are:

- Minimizing the necessity for and amount of personnel time spent in radiation areas
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment that require personnel attention

The BWRX-300 is designed to ensure the radiation exposure to any nuclear energy worker is less than the limits specified in the SOR/2000-203 (Reference 12.3-2).

##### **12.3.1.1 Radiation Source Term Reduction**

Radiation source term reduction design considerations:

1. N-16 skyshine dose contribution is determined from N-16 activity in the vessel nozzle outlet steam based upon the hydrogen water chemistry.
2. Noble gas airborne concentrations are negligible throughout the TB except for inside the SJAE cubicles. The process air from these cubicles is exhausted to the environment after HEPA filtration and noble gas decay in activated carbon beds in the gaseous RW system reducing contamination in adjoining areas.
3. Minimizing airborne radioactivity includes:
  - a. Storing underwater the steam dryer, separator, and chimney during refueling operations
  - b. Cooling the RB pools with large heat capacity heat exchangers
  - c. Sweeping the spent fuel pool surface and preventing evaporative pool losses and gases from mixing with the area atmosphere using the FPC
4. Radioactive material is kept in containers to the extent possible.
5. Equipment cubicles, corridors, and areas routinely occupied by operating personnel do not contain significant airborne radioactivity sources. Radioactive equipment that potentially leaks is installed in separate shielded compartments.
6. Material selection minimizes the creation of activated corrosion products.
7. Zinc injection minimizes dose rate from Co-60 deposition.

#### **12.3.2 ALARA Design Guidelines for Mechanical System and Components**

This section describes the incorporation of the ALARA principle in the BWRX-300 systems and components design complying with REGDOC-2.7.1, Section 4.1 ALARA (Reference 12.3-1).

##### **12.3.2.1 General Mechanical Design Guidelines**

1. Shielding, remotely operated valves, and sample transporting casks reduce radiation exposure.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

2. Equipment is designed for normal and abnormal operating conditions while maintaining radiation exposures ALARA.
3. Passive equipment is used where possible, thus minimizing worker exposure.
4. Process equipment that accumulates radiation sources from filtering process streams, such as the condensate and feedwater and the fuel pool cooling systems, are remotely operated, including backwashing operations.
5. Equipment facilitates maintenance.
6. Refueling tools have smooth surfaces and are easily drained to reduce contamination.
7. Vessel and piping insulation is easily removed and restored.
8. Equipment and piping minimize crud buildup and provide easy decontamination and maintenance.
9. Remotely operated mechanical devices are used to inspect the RPV body and nozzle welds.
10. In-Service Inspection (ISI) laydown space accommodates insulation removal.
11. These services are in close proximity to equipment facilitating maintenance and inspection:
  - a. Electrical outlets
  - b. Welding outlets
  - c. Condensate grade and/or demineralized water
  - d. Compressed air
  - e. Communications
12. For areas where the dose rate is 50  $\mu\text{Sv/hr}$  (5 mrem/hr) or higher where worker access is limited due to the radiation zone (as described in Section 12.6), or where equipment is removed for maintenance, the equipment design provides rapid removal using “plug in” or “quick disconnect” piping connections, electrical leads, instrumentation connections, and support linkages.
13. Passageways are provided for access ease where equipment is removed and replaced for maintenance by plant personnel wearing full Personnel Protective Equipment (PPE).

**12.3.2.1.1 Mechanical Components**

**Tanks**

1. Overflow pipes are connected to liquid radioactive waste drain system for processing. Vents run to the building ventilation exhaust system.
2. To facilitate decontamination, corrosion-resistant materials and linings prevent corrosion products buildup on the walls.
3. Walls or curbs and drains to the liquid radioactive waste system are provided around the tank for leakage collection. Room collection capacity is sized to contain the tank volume.
4. Spent resin, phase separator, and backwash tanks are located in enclosures free of flammable or combustible substances.
5. Bottom valves on slurry tanks are flush-mounted.



### **Heat Exchangers in Radioactive Service**

1. Tube and shell heat exchangers are designed with excess tubes accommodating tube plugging.
2. Where feasible, leakage from highly contaminated fluids is prevented from entering clean fluids by maintaining a higher pressure in the clean fluid sided.
3. Heat exchanger design allows for fluid drainage avoiding pooling that may lead to radioactive crud deposition.
4. Heat exchanger drains are hard piped to floor drains.
5. Cleaning connections are provided for condensate, demineralized water, and/or chemical solutions.

### **Piping**

1. Piping selection provides a service life equivalent to the design life of the plant, with corrosion and environmental condition allowances.
2. Contaminated piping systems are welded to the extent possible.
3. Piping in highly radioactive systems such as the spent fuel pool cooling system have butt-welded connections, rather than socket welds, reducing crud traps.
4. Radioactive piping is sized to produce turbulent flow to maintain solids suspension without plugging.
5. In systems where contamination settling can occur, the drains are placed on the side of the piping.
6. Systems that contain multiple pumps are placed in separate alcoves with piping routed into shielded pipe chases.
7. Radioactive piping minimizes plant personnel radiation exposure as follows:
  - a. Radioactive pipe routing in corridors is minimized
  - b. Shielded pipe trenches and pipe chases are used for routing high-activity pipes in low-level radiation areas
  - c. Separating radioactive and non-radioactive pipe routes for maintenance
  - d. Piping is arranged based upon SC, grouping of components, system operability, and maintenance access
  - e. Non-radioactive line systems are not run inside biologically shielded compartments except where needed to supply services inside the compartment
  - f. Piping configurations minimize the number of "dead legs" and low points to avoid radioactive crud and fluids accumulation
8. Systems are designed with vents and flushing lines reducing crud buildup.
9. Instrument lines, except those for the reactor vessel, are designed for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines are flushed with condensate from the instrument rack when the reactor is depressurized.
10. Piping conveying highly contaminated fluids is routed through shielded pipe chases and shielded cubicles. When these options are not feasible, the radioactive pressurized

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

process piping is sleeved embedded in concrete walls and floors. Embedded pipes connected to floor drains are butt-welded stainless steel.

11. Contaminated piping design facilitates cleaning and removal during decommissioning.
12. "Clean" services, such as compressed air and demineralized water are routed through shielded pipe chases.
13. Where piping penetrates walls the use of embedded piping is minimized facilitating system dismantling and plant decommissioning.
14. When radioactive and non-radioactive pipes are collocated, drain provisions are provided for removing radioactive fluids, and the valves are remotely controlled.
15. Piping penetrations through shield walls minimize the shine on surrounding areas.
16. Underground piping is installed for clean service only. If not possible, it is routed with a guard pipe, trench or tunnel for visual inspection and monitored leakage. Threaded and flanged connections are kept to a minimum. Other joints are welded or permanently bonded depending on the piping material.

#### **Ion Exchange and Activated Carbon Beds and Filters**

1. Filters in purification and cleaning systems are operated and cleaned remotely.
2. For purification systems, preparation, and resin changeout are performed remotely.
3. A mesh trapping resins or activated carbon particles is installed downstream of vessels.
4. Sampling points are installed at the inlet and outlet of, ion exchange vessels and activated carbon beds.
5. Cartridge-type filters or duplex filters with on-line exchangers are sized for one fuel cycle without changeout.

#### **Pumps and Valves in Radioactive Service**

1. Pumps located in radiation areas are designed with features to minimize maintenance time: quick-change cartridge-type seals, back pullout features that permit removal of the pump impeller, or mechanical seals without disassembly of attached piping.
2. Where two or more pumps conveying highly radioactive fluids are located adjacently, shielding is provided between them maintaining exposure levels ALARA. Pumps adjacent to other highly radioactive equipment are shielded to reduce personnel exposure.
3. Pump control instrumentation for process equipment is located outside high radiation areas in separate alcoves. Motor or pneumatic-operated valves or valve extension stems are used allowing operation.
4. Radioactive system pumps and valves are remotely controlled.
5. Pump casing drains are provided on radioactive systems and drained to the liquid radioactive waste system for processing.
6. Valves have cartridge-type packing that is easily replaced, or diaphragm valves or quarter-turn ball valves are used that do not require similar packing.
7. Instrumentation and valves in high radiation areas are remotely operated.
8. Straight-through valves are selected over those that exhibit flow discontinuities or internal crevices minimizing crud trapping.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

9. Valves are placed in horizontal runs to facilitate dismantling.
10. For systems in high radiation areas, gasket materials are selected that inhibit degradation. Degradation mechanisms (i.e., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue etc.) are considered during the material selection process.
11. Valves containing radioactive fluids are separated from “clean” services to reduce the radiation exposure from adjacent valves and piping.
12. Safety and relief valves and depressurization valves are designed with flange connections to allow whole valve removal.
13. Isolation valves are located so that fluid contained in dead branch-pipes is minimized.
14. Back seat valves are used reducing leaks through the packing when the valve is full open.
15. Pipe insulation around valves is easily removed.

### **Radioactive Drains**

1. For systems containing highly radioactive fluids, frequently used drains are hard piped to the nearest drain. An air gap exists between the system drain and the floor or equipment drain hub. For drains used less frequently, a drain connection with a valve is provided so plant personnel can connect a hose and direct the outflow to the nearest drain.
2. Sump vents are located near the room exhaust vent register to control airborne radioactivity released from sump discharges.
3. Sumps are stainless steel-lined reducing corrosion and providing easily decontaminated surfaces.
4. Pipes carrying fluid to radioactive drains and clean drains are separate and independent.
5. The high-activity level resin slurry drains are separated from other drains and the contents routed to liquid radioactive waste for subsequent treatment.
6. Drain lines are gravity-sloped.
7. Drain lines containing highly radioactive fluids are routed through pipe chases or shielded cubicles.
8. One piece floor drains minimizing possibility of liquid penetrating at embedment boundaries. Non-porous alternatives to grout such as epoxy are used for the installation of floor drains.
9. Drain lines contain traps that prevent the flow of radioactive gas from compartment to compartment.
10. The drains from rooms that house spent resin tanks, phase separator tanks, or demineralizer vessels are equipped with normally closed isolation valves.
11. Sumps are located at the lowest level of the buildings.

### **Sludge and Resin Systems**

System design and implementation prevent accumulating and retaining radioactive sludge and resins in transfer pipes by:

1. Pipe runs are reduced.
2. Low points and blind sections are avoided.
3. Appropriate pipe slope to support gravity flow is used.

4. Pipe flow restrictions are avoided.
5. Short 90° elbows are avoided and replaced by large-radius curved sections at least five times the diameter, or two 45° elbows spaced apart.
6. Whenever T connections are required, the main flow runs through the straight section. If the secondary flow is into the T, then the connection is made above the axis of the pipe. If the secondary flow is out of the T, then the connection is on the lateral connector side.

#### **12.3.2.1.2 Materials**

1. The reactor coolant system materials selected prevent corrosion product formation. Reactor internal components, except for the zircaloy in the reactor core, are stress corrosion-resistant stainless steels or other high alloy steels.
2. Carbon steel used in systems processing or storing reactor coolant or steam are low in nickel content and contain small amounts of cobalt impurity.
3. Main condenser tubes and closed loop cooling heat exchangers cooled by the plant cooling water tubes and their tube sheets are low-cobalt stainless steel or titanium alloys and low-cobalt materials.
4. Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750 that have high nickel content, are used in some reactor vessel internal components. These materials are used in applications where special requirements (possessing specific thermal expansion characteristics along with adequate corrosion resistance) are necessary, and where no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.
5. The BWRX-300 limits the use of cobalt-bearing materials on moving components that are historically identified as major sources of reactor coolant contamination. Stellite is used for hard facing of components that must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. Alternative materials (Colmonoy® and NOREM®) are used for some hard facings.
6. Control Blade Materials: All control rod materials are less than 0.03 weight percent cobalt. The average niobium content for the handle and absorber section, less boron carbide and hafnium, is less than 0.1 weight percent.
7. Non-stainless steel equipment has an adequate finish and is protected with a non-corrosive material aiding in decontamination for some applications.
8. For systems in contact with reactor coolant (pumps, valves, etc.), component materials are as degradation-mechanism resistant as possible. Degradation mechanisms (i.e., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue etc.) are considered during the material selection process.
9. System components are made of materials that are qualified to withstand pressure, temperature, and radiation minimizing maintenance.
10. Pipes are designed for a 60-year life, based on the environmental conditions and corrosion preventing maintenance, unless documented otherwise (e.g., abrasive slurry lines). Pipes with less than a 60-year life are replaceable (i.e., not embedded in concrete).
11. Long-life materials that withstand the operating pressure, temperature and radiation are used for component joints and seals minimizing maintenance.

12. Heat exchangers are constructed of stainless steel or titanium tubes minimizing failure and reducing maintenance requirements.

### **12.3.2.2 Mechanical Systems, Equipment and Components Arrangements**

#### **12.3.2.2.1 Confinement of Systems**

1. Radioactive systems are isolated from non-radioactive systems by confining them in different enclosures. Adequate shielding is provided between both areas.
2. Clean systems (non-radioactive) are in areas with no radioactivity or with low background radioactivity levels.
3. Piping and pumps or systems that have components that are major sources of radiation are in separate cubicles to reduce exposure during maintenance.
4. The various redundancies of a radioactive system that are provided for the same function are in separate areas and shielded from each other.
5. As a consequence of normal steam and water leakage into containment, airborne radioactive contamination exists during normal operation. Containment atmosphere purging to the environment is via the containment purge system, that is routed and processed through HEPA filters then to the TB plant stack.

#### **12.3.2.2.2 Confinement of Equipment**

1. Radioactive system equipment with different radiation levels are in different rooms.
2. System equipment that accumulates or contains radioactive fluids (tanks, heat exchangers) are isolated from other equipment (pumps, valves) and located in different rooms.
3. System equipment that concentrates radionuclides (particulate filters, and demineralizers) are isolated from the rest of the equipment in the system and located in different rooms or shielded.

#### **12.3.2.2.3 Location of Components**

1. Radioactive piping is accessible and arranged allowing inspection and maintenance with minimum exposure to personnel.
2. The location of equipment is optimized to minimize the length of the radioactive fluid ducts between them.
3. Functionally different radioactive system components such as control valves and pumps that require frequent maintenance or inspection, are in different rooms, to avoid personnel exposure.
4. Radioactive equipment is located as far as practicable from personnel dwelling and passage areas.
5. Labyrinths are used in radioactive equipment room accesses so that personnel are not exposed directly in corridors or staircases. Double labyrinths are used to shield high intensity radiation sources. Sealed cubicles with hatches or shield doors are used to shield very high radiation sources.
6. Intense radiation source equipment rooms are arranged so that workers are not exposed to the source while in the room.
7. Rooms house piping associated with the functional system.

8. Radioactive pipes are run inside trenches or shielded pipe boxes whenever the run is through a low-level radiation area.
9. Radioactive pipes are grouped on the opposite side of the entrance to the room and away from components that require frequent maintenance.
10. Radioactive pipes are sloped unless there are drains or purges at low points.
11. Equipment startup, handling devices, and actuation mechanisms are in low activity areas.
12. Sampling equipment, instrumentation, and stations requiring routine attention (maintenance, calibration, actuation, reading or inspection) are in places that are easily and quickly accessible (next to the entrance to the enclosure) and in low activity areas.

#### **12.3.2.2.3 Auxiliary Spaces**

1. Remotely operated, back-flushable filter systems are provided for radioactive RW and cleanup systems.
2. The following auxiliary spaces prevent contamination dispersion:
  - a. Posted areas for storage of contaminated parts and equipment before they become decontaminated
  - b. Areas where equipment is decontaminated before repair. (e.g., the PLSA maintenance rooms that may be used for disassembling pumps, motors, etc.)
3. Decontamination areas have condensate quality water available to carry out preliminary decontamination of equipment segregated for repair.

#### **12.3.2.3 Minimization of Crud Buildup**

The crud buildup minimization facilitates decontamination and cleaning equipment, components, and contaminated areas, and minimizes airborne contamination dispersion.

Flushing provisions are made for equipment for minimizing crud buildup.

### **12.3.3 In-Service Maintenance and Inspection Provisions**

#### **12.3.3.1 Layout of Components, Space and Access Provisions**

The following provisions prevent or minimize radiation exposure:

1. Adequate means are provided to monitor and control access to medium and high-activity areas (These are radiation zones C to I, inclusive as described in 12.6).
2. Equipment and component laydown areas have adequate space for dismantling, repair, removal, and insulation restoration for inspection or maintenance. Adequate access is also provided for transporting failed and replacement components and maintenance tools.
3. The welds in the reactor cooling circuit and in the pipes connected to it requiring ISI are readily accessible.
4. Sufficient space is provided to store and use temporary shielding.

#### **12.3.3.2 Staircase and Platform Requirements**

The following design requirements minimize time spent in radiation areas:

1. Stairs are provided and stepladders or ladders are avoided.
2. Stairs are sized for wearing protective gear and carrying necessary tools.

3. Platforms and staircases used to access components requiring frequent inspection or maintenance are permanent structures sized with sufficient space and capacity to withstand the component size and weight and additional mobile shielding.

#### **12.3.3.3 Hoisting and Transport Mechanisms**

The following provisions limit the operations and exposure duration:

1. Space is available for installing and hoisting transport mechanisms such as cranes, hoists, monorails, hand rigging, and trolleys used during periodic inspection and maintenance. Monorail beams are permanently installed over heavy equipment (equipment weighing one metric ton or more).
2. Hoisting/transport mechanisms are provided to handle heavy equipment, component parts, slabs, and heavy removable structural elements.
3. Hoisting/transport mechanisms are provided to handle parts and components in decontamination areas, in contaminated storage areas and active workshops
4. Remotely operated electric hoists are used for tasks/equipment associated with general area dose rates in excess of 1 mSv/hr (100 mrem/hr) where access is limited to one hour per week (In alignment with radiation zone E as described in 12.6).
5. Embedments are provided, in walls, floors, and overhead facilities for special rigging that requires scheduled maintenance. The design load capacity is marked on each embedment.

#### **12.3.4 Room Leak-Tightness and Drain Provisions**

The following measures prevent contamination dispersion and radiation passing through holes in walls and structures:

##### **Leak-Tightness**

1. House equipment enclosures with large radioactive liquid volumes (tanks, heat exchangers, filters, etc.) are leak tight. Additionally, containment measures are provided to prevent dispersion in the event of liquid overflow (curbs, sloping floor drains).
2. Gaseous radioactive waste system is leak tight minimizing radioactive gases and potentially flammable hydrogen escape.
3. Penetrations in walls and structures separating enclosures and buildings are located and sealed to prevent fluid contamination if a break or leak develops.

##### **Drains**

1. Enclosures that contain equipment used to store radioactive liquids have drains for spills or failures.
2. Sample station effluents are returned to the process stream or to the liquid radioactive waste through a common return line.

#### **12.3.5 General Considerations of Systems that Process Radioactive Fluids**

##### **12.3.5.1 System Isolation**

1. When a contaminated or potentially contaminated system is connected to a clean (i.e., non-contaminated) system, provisions are implemented ensuring backflow cannot occur. Backflow prevention is provided by double isolation and double check valves that permit periodic sampling ensuring clean systems are not contaminated.

2. All contaminated or potentially contaminated systems are isolated from outside environment or otherwise equipped with isolation devices to control discharges.
3. Systems discharges to the environment are equipped with radiation monitoring devices to automatically isolate the discharge if high activity is detected. The high-activity setpoints for these systems are provided in the POSAR that ensure compliance to the (Darlington New Nuclear Project (DNNP) site derived release limits.

#### **12.3.5.2 Sealing of Radioactive Components and Leak Isolation**

1. Gaseous radioactive systems are sealed.
2. Equipment designs includes leak isolation, limiting leak devices, or controlling fluid leaks by piping to the sumps or using drip pans with drains piped to the floor drains.

#### **12.3.5.3 Corrosion Limitation Products**

1. Thermal cycle system corrosion is limited by using high quality, corrosion-resistant materials, and continuously controlling other properties via water chemistry.
2. Retention and elimination of corrosion products with the use of CFD are incorporated into the thermal cycle systems.
3. Features preventing flow discontinuities that lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout. Mitigating engineering features are added where avoidance is not possible.
4. Backing rings and consumable inserts are not used in pipes containing radioactive materials.

#### **12.3.5.4 Activity Control and Monitoring Systems**

The activity control and monitoring criteria are described by system function.

##### **12.3.5.4.1 Process Sampling System**

Sample stations in the plant provide routine reactor water quality surveillance:

1. Local sampling points and sampling stations are in low radiation areas.
2. Fume hoods are used for airborne contamination control. Sample sinks, fume hoods, and glove boxes are constructed of polished stainless steel and/or plastic easing decontamination if a spill occurs.
3. Grab spouts are located above the sink reducing contaminating surrounding areas during the sampling process.
4. Radiation shielding is used for panels, local sampling boxes, cells, and racks to maintaining ALARA.
5. Sampling stations are closed system and the grab samples taken at the sampling stations have a chemical fume hood preventing operating personnel exposure. Constant air velocity is maintained through the working face of the hoods ensuring airborne contamination does not escape to the room under operating conditions.
6. A shut-off valve is placed in the sample lines immediately downstream of the sample nozzle, and sampling lines not continuously used during normal operations are drained-sloped to prevent crud pockets or areas of stagnant fluids.



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

7. If a sample cannot be returned to its original system, it is directed to the nearest appropriate drain.
8. Provisions are made for flushing pipes with demineralized water before the sample bottle is removed from the panel. Panel drains are drained to the appropriate drain system.
9. Means are provided for remote handling and shielded transport of sample bottles, when required.
10. Connections to containment airspace are provided facilitating atmospheric sampling for isotopic composition and hydrogen and oxygens concentrations in the event of an accident.

**12.3.5.4.2 Process Radiation and Environmental Monitoring System**

1. Gross beta-gamma activity is constantly monitored in the main steam line to detect potential increases in fuel leaks to the coolant.
2. Activity monitors are installed close to the interface barriers where clean fluids that serve to cool the contaminated process fluids might receive radioactive fluid leakage.
3. Activity monitors are installed to monitor and control system discharge. These systems are isolated upon high level detection.
4. Area radiation monitors are installed in:
  - a. Areas where doses may significantly increase either periodically or inadvertently
  - b. Areas where unauthorized or inadvertent movement of radioactive material may occur
  - c. Areas where leaks of radioactive material from process streams may occur
  - d. Areas where nuclear fuel is stored or handled, such as areas close to the refueling machine and close to the fuel pool
  - e. Areas frequently accessed for operation or maintenance requirements

To prevent radiation overexposure, to detect leaks, and to prevent high-activity discharge, the atmospheres of the rooms and stacks are monitored for particulates, noble gases, and iodine. Radiation monitor locations and descriptions for the plant are provided in Subsection 12.3.14 Monitoring of Individuals and Working Areas (Instrumentation for Radiation Levels and Airborne Radioactivity).

The Containment, TB, TB, PLSA, and RWB atmospheres are monitored for particulates, noble gases, and iodine:

- Laboratories are monitored continuously for noble gas and particulate activity levels, and periodically for iodine
- Atmosphere adjacent to personnel access hatches is monitored for particulates
- Buildings housing large equipment that store reactor coolant are continuously monitored for noble gases and particulates
- Buildings where radioactive waste is stored are continuously monitored for particulates
- Refueling machine atmosphere is continuously monitored for noble gases during refueling

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

- Particulates, via an iso-kinetic sampler, and noble gas activity levels are continuously monitored in the TB stack during normal and shutdown operations, and periodic surveillance

Activity monitoring systems detect and signal off-normal operating conditions for:

- High differential pressure in the sampling filters
- High and low air flow in the activity monitoring lines
- Malfunction of a blower or monitor
- Incorrect operation of valves

### **12.3.6 ALARA Structures Criteria**

This section delineates the main ALARA criteria provided in the BWRX-300 structural design.

#### **12.3.6.1 Building/Structure Specific Shielding Requirements**

1. Containment - The major shielding structures consist of containment walls and reactor shield wall (bioshield):
  - a. The reactor shield wall consists of shielding using the steel bricks and steel plate.
  - b. The containment outer wall is a steel concrete steel cylinder or steel bricks, that totally surrounds containment. A steel brick top slab covers containment.
  - c. The shielding for the RB maintains areas requiring regular daily access to dose rates less than 6  $\mu\text{Sv/hr}$  (0.6 mrem/hr).
  - d. All necessary shielding in the upper containment maintains dose ALARA during transfer of irradiated spent fuel assemblies.
  - e. Shielding or appropriate controls minimize dose rates if a fuel handling mishap results in dropping a fuel assembly across the reactor flange.
2. Penetrations through the containment wall are shielded reducing radiation streaming. Localized dose rates outside these penetrations are limited to less than 50  $\mu\text{Sv/hr}$  (5 mrem/hr). The penetrations through interior shield walls of the RB are shielded using a modified high-density silicone elastomer, or equivalent absorber, reducing radiation streaming. Penetrations are also located to minimize radiation streaming consequences into surrounding areas.
3. The CUW RHX is located in a shielded cubicle.
4. Demineralizer and filter units are in separate shielded cubicles. This arrangement allows maintenance of one unit while the other operates. The dose rate in the adjoining demineralizer cubicle from the operating unit is less than 250  $\mu\text{Sv/hr}$  (25 mrem/hr). Infrequent entry into the demineralizer cubicle is through shielded hatches. The bulk of the piping and valves for the demineralizers and filters are in an adjacent shielded valve gallery.

Backflushing and resin application of the demineralizers are controlled from an area where dose rates are less than 10  $\mu\text{Sv/hr}$  (1 mrem/hr).
5. Fuel Storage - The fuel storage pool ensures the dose rate around the pool area is less than 25  $\mu\text{Sv/hr}$  (2.5 mrem/hr).
6. Fuel Handling - Integral shielding installed on the refueling machine is equivalent to one foot of water. Refueling pool depth maintains less than 25  $\mu\text{Sv/hr}$  (2.5 mrem/hr) at the

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

water surface (per ANSI/ANS 57.1) during transit of a single grappled fuel bundle from/to the reactor vessel and the spent fuel racks.

7. The dose rate in the Main Control Room (MCR) is limited to a maximum threshold of 2  $\mu\text{Sv/hr}$  (0.2 mrem/hr) during normal reactor operating conditions. The outer walls of the CB are designed to attenuate radiation from within the RB and from possible airborne radiation surrounding the CB following beyond design basis event. The wall shielding limits the direct-shine exposure from all potential high radiation sources.
8. Noble gas airborne concentrations are expected to be negligible throughout the TB except near the SJAES. These components are placed in shielded cubicles that are not normally occupied during operation.
9. Facilities supporting responses to accidents and emergencies in accordance with REGDOC-2.5.2, Section 8.13.3 and REGDOC-2.10.1, Sections 5.2 and 5.7 are provided with radiological protection and monitoring equipment necessary to ensure that personnel radiation exposure during accident recovery does not exceed 50 mSv (5 rem).

#### **12.3.6.2 General Layout and Accesses**

1. Radiological protection layout follows the concept of separation into three distinct and isolable areas: uncontrolled access areas, controlled (limited) access areas, and uncontrolled areas.
2. The controlled access areas encompass all radioactive systems, equipment, components, and materials.
3. The buildings and areas that define the controlled access area are separated and/or have sufficient shielding to maintain dose rates in uncontrolled areas to less than 6  $\mu\text{Sv/hr}$  (0.6 mrem/hr.)
4. The routes between buildings of the controlled access area and their accessible rooms are short as possible, designed for easy passage, and run through low radiation areas.
5. All accessible rooms are easily reached from service corridors.
6. The accessible areas floor-ceiling height accommodates personnel transporting tools for inspections and maintenance.
7. Areas are provided for storing removable components and mobile shielding that do not hinder personnel passage and mobility.
8. Plant areas are arranged for service operations, monitoring or inspection of highly radioactive equipment that is carried out manually or remotely.
9. Easy escape routes are provided with appropriate signage posted.
10. Access to high radiation areas is posted and blocked by physical barriers.
11. Access to enclosures accommodates entry and exit of components and tools.
12. Properly sized access hatches are provided (e.g., installation or removal of plant components).
13. Equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, are located for ease of access and minimum occupancy time.
14. Highly radioactive equipment plant areas are laid out to allow remote or mechanical operation, service, monitoring, or inspection.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

15. Transportation equipment (monorails, carts, hoists, etc.) is provided so that equipment is serviced in lower radiation areas.
16. Centralized control panels are provided permitting remote operation of all SC equipment from the lowest radiation zone practicable and are provided for SCN equipment.
17. Adequate space is provided for movable shielding for sources within the service area.
18. Means to control contamination and facilitate decontamination of contaminated areas are provided.
19. Space for pumps and valves not containing radioactive fluids are outside radioactive areas.
20. Pump configuration provides sufficient space for maintenance, including laydown space for disassembled piping, insulation, and components and for maintenance support features such as monorails and work platforms.
21. Personnel exposure is minimized during inspection and maintenance by locating equipment and instrumentation as far as possible from potential sources of high radiation.
22. The BWRX-300 plant does not contain underground pipelines that are directly buried in the ground (i.e., contained in concrete trenches/tunnels or concrete duct bank) or buried in masonry wall construction.
23. Penetrations through outer walls of the building containing radiation sources are sealed to prevent miscellaneous leaks into the environment.
24. Sloped floor drains are provided in shielded cubicles and other areas where spill potential exists limiting contamination.
25. Smooth, flat surfaces are used where radioactive spills could occur and expanded metal-type floor gratings are minimized.
26. The RW tank cubicles concrete is provided with a sealant and tank cubicle stainless steel liners to prevent any water releases to the environment.
27. The RB washdown bays are designed so that after the spent fuel cask is loaded on the transporter, potential surface contamination is monitored and decontaminated.
28. The washdown bays include the following:
  - a. Walls or curbs are located around areas of potential contaminated fluid leakage
  - b. Floor surfaces are sloped to drains, and sumps sized for maximum decontamination liquid flow rate
  - c. Concrete surfaces, including floor surfaces are protected with a non-porous coating
  - d. The decontamination fluid is processed through the liquid RW system
29. Health physics facilities and features administratively control:
  - a. Plant personnel activities limiting personnel exposure to radiation and radioactive materials ALARA, and within the guidelines of REGDOC-2.7.1
  - b. Effluent releases to maintain releases ALARA, and within the limits of CSA-N288.1, and the plant Operational Limits and Conditions (OLC)
  - c. Waste shipments meet applicable shipping and receipt material requirements at the storage or burial site

### **12.3.6.3 Walls, Structures and Shielding Elements**

1. The shielding requirements:
  - a. Limit radiation exposure to the public, workers, and the environment that are ALARA and comply with REGDOC-2.7.1, Section 4.4.1 (Reference 12.3-1) requirements
  - b. Limit radiation exposure to workers, in the event of an accident, to levels that are ALARA and consider the limits in ICRP Publication 103 (Reference 12.3-3)
  - c. Limit the radiation exposure of critical components assuring component performance and design life are not impaired
2. Shielding reduces radiation levels from N-16 sources in occupied areas during power operation.
3. Shine between joints of different shielding elements, especially removable blocks, are avoided preventing radiation passage.
4. Penetrations through shield walls are avoided reducing the number of streaming paths. Whenever penetrations are required through shield walls, they are located minimizing the effect on surrounding areas. Penetrations are located so that the radiation source cannot pass through the penetration. When this is not possible, an added order of radiation source reduction is provided.
5. The annular area between pipe and penetration sleeves and electrical penetrations are filled with shielding material reducing the streaming. Examples of the shielding materials used in these applications include modified high-density silicone elastomer (or equivalent), with a density comparable to concrete, and boron-loaded refractory-type material for applications requiring neutron as well as gamma shielding.
6. Whenever a penetration cannot be avoided and a simpler and more effective solution does not prevent the radiation passing through the penetration, a shielded enclosure is installed around the penetration as it exits the shield wall.
7. The maximum number of elements is fed through penetrations and openings minimizing the number of openings.
8. The number of openings between highly radioactive areas and accessible areas are minimized.
9. Equipment is located minimizing shielding requirements. Shielded labyrinth entrances are selected for radioactive pump, equipment, and valve rooms instead of shield doors and hatches. Adequate space is provided in labyrinth entrances for access.
10. All systems containing radioactivity are identified and shielded based on access and exposure level requirements for surrounding areas.
11. Provisions are made for shielding major radiation sources during ISI to reduce inspection personnel exposure. For example, steel platforms are provided for ISI of the RPV welds and associated piping.
12. Concrete used for shielding purposes is designed considering guidance in Regulatory Guide 1.69 (Reference 12.3-4). Where special circumstances dictate, steel, encapsulated lead, water, a modified high-density silicone elastomer, or a boron-laced refractory material is used.

13. Shielding thicknesses are selected reducing the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps.

#### **12.3.6.4 Pools Containing Radioactive Liquids**

1. The liner welds for all pools are equipped with a leak detection system.
2. The leak chase is grouped to different pool areas and leakage is directed to area drains. This allows both leak detection and determining where leaks originate. After construction is finished, each isolated pool is leak tested.

#### **12.3.6.5 Doors and Shield Walls**

1. Doors are installed to manage access to high radiation level areas. Additional measures such as radioactive signs or locks may be used.
2. The doors in shielding labyrinths are located at the entrance to the labyrinth.

#### **12.3.6.6 Coatings, Surface Finishes and Painting**

1. Epoxy-type wall and floor coverings are used providing smooth surfaces easing the decontamination. Smooth surfaces are applied to the walls and floors of the RB, TB, RWB, and TB Annex rooms containing equipment with liquid radioactive sources, floor drain areas, and washdown bays.
2. A non-porous surface paint and finish for easing decontamination is applied to room walls, floors, staircases, and platforms.
3. Protective coatings are qualified and capable of surviving a Design Basis Accident (DBA) without adversely affecting safety mitigating SSCs per guidance in Regulatory Guide 1.54 (Reference 12.3-5).
4. A contrasting undercoat color is included in the coating system for wear indication.

#### **12.3.7 ALARA Criteria for Electrical Instrumentation and Control Design**

The ALARA criteria are applied to the electrical and instrumentation design.

##### **12.3.7.1 Component Arrangement**

1. Motor control centers and system control panels are in low radiation level areas.
2. Motor-operated valves are controlled from low radiation level areas.
3. The cable trays and equipment layout does not hinder personnel passage.
4. The electrical motors connectors located in high radiation zones has quick disconnects simplifying repair and maintenance.
5. The RW piping gallery between the TB and the RW contains only SCN electrical cables that are separated from the RW piping by a minimum shielding of 20 cm (7.9 in).

##### **12.3.7.2 Lighting, Telephone and Loudspeaker System**

1. Lighting levels are suitable for surveillance and maintenance operations in radiation areas.
2. Multiple lighting fixtures prevent the immediate need to replace faulty bulbs in shielded cubicles.
3. Lighting fixtures are in easily accessible locations reducing exposure time during bulb replacement.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

4. Light bulbs within containment, main steam tunnel, and refueling level of the RB containing mercury are not used due to mercury's neutron activation potential.
5. Radioactive areas rooms and corridors have emergency lighting and illuminated exit signs to indicate escape routes.
6. Alarms and the personal address system use uninterruptible power supplies.
7. The lighting equipment, communications, and loud-speakers location allows for maintenance with minimal radiation exposure.

#### **12.3.7.3 Instrumentation**

1. Instrumentation for equipment operation, control, and sampling is located in low-level areas (shielded rooms for valves, corridors, or control rooms) that are accessible during normal and off-normal operating conditions. Shielded valve galleries are provided for the CUW, FPC, and RW (e.g., spent resin tank) systems.
2. Sensing lines are routed from primary system taps avoiding placing transmitters or readout devices in high radiation areas. For example, reactor water level sensing instruments are located outside containment.
3. Instrument sensors requiring inspection, repair, or maintenance accessibility are located facilitating operations.
4. Instruments located in high radiation areas allow removal to low radiation areas for maintenance.
5. Transmitters and display instruments are in radiation zones A or B to the extent possible. The radiation levels in these zones are defined in Subsection 12.6.1.
6. Sampling instrumentation system design minimizes the required amount of radioactive fluid.
7. Radiation monitors located in high radiation areas have displays located in an adjacent low radiation areas and local visible and audible alarms.

#### **12.3.7.4 Materials**

Components and cables used in radioactive areas are qualified for the calculated radiation conditions.

### **12.3.8 System and Major Component Design Considerations**

#### **12.3.8.1 Main Condenser**

1. Main condenser hotwell provides a holdup for Main Steam Isolation Valve leakage.
2. During normal operation, the main condenser shells operate with a vacuum. Radioactive circulating water leakage to the atmosphere does not occur. Any air leakage is into the main condenser shell side.
3. Condensate is retained in the main condenser permitting radioactive decay before entering the condensate system.
4. Necessary shielding and controlled access for the main condenser is provided.

#### **12.3.8.2 Nuclear Boiler System**

1. Entry into the RB steam tunnel is through a controlled and shielded personnel access door that has a concrete labyrinth to attenuate radiation streaming from the adjoining steam lines. During reactor operation, the steam tunnel is controlled access.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

2. Selected valves within the steam tunnel are provided with drains that are piped to drain sumps minimizing leakage into surrounding areas. Floor drains are provided within the steam tunnel minimizing contamination if a leak occurs.
3. Penetrations through the steam tunnel walls are minimized, reducing streaming paths. Penetrations through the steam tunnel walls are located exiting controlled access areas or in areas not aligned with the steam lines. A modified high-density silicone elastomer (or equivalent) is employed whenever possible for these penetrations reducing the available streaming area presented.
4. The steam tunnel shields the plant complex from N-16 gamma shine in the main steam lines. The steam tunnel walls provide shielding limiting the direct-shine exposure from the main steam lines at any point that are inhabited in areas adjacent to the steam tunnel to less than 50  $\mu\text{Sv/hr}$  (5 mrem/hr) during normal operations (controlled access area).

#### **12.3.8.3 Fuel Pool Cooling System**

1. The FPC system services the fuel pool, cask pit, and reactor cavity pool on a rotating basis. This system operates continuously removing generated heat, reducing pool evaporation, and reducing contamination.
2. The FPC system includes two independent filters and demineralizers that remove radioactive contamination from the fuel pool, cask pit, and reactor cavity pool. These units are the highest radiation level components in the system. Each unit is located in a concrete-shielded cubicle accessible through a shielded hatch.
3. Remotely backflushing the filters and demineralizers is possible. FPC system filters are backwashed into a receiving tank and routed to the liquid radioactive waste system.
4. Clean water connection flushing lines are provided for the FPC system prior to switching between pools preventing contamination between pools.
5. All processing FPC system valves (inlet, outlet, recycle, vent, and drain) on the filters and demineralizers are located outside the shielded cubicles in a separate shielded cubicle or area together with associated piping, headers, and instrumentation.
6. FPC system piping containing resin is sloped downward to the receiving system or tank.
7. All shielded FPC system components are consolidated in the same section of the RB. Personnel access to shielded system components is controlled. Shielding for the components reduces the radiation level to less than 10 mSv/hr (1000 mrem/hr) in adjacent areas where normal access is permitted.
8. FPC system operation is accomplished from the MCR, and local control panels located where radiation levels are less than 50  $\mu\text{Sv/hr}$  (5.0 mrem/hr), and controlled limited access is permitted.
9. The FPC system provides for the collecting, monitoring, and draining pool liner leaks from the fuel pool, cask pit, and reactor cavity pools to the Liquid Waste Management System.
10. The fuel pool is equipped with drainage paths for:
  - a. Preventing stagnant water buildup behind the liner plate
  - b. Preventing the uncontrolled loss of contaminated pool water
  - c. Providing liner leak detection and measurement



11. The fuel pool, cask pit, and reactor cavity pool is also equipped with stainless steel liners and leak detection drains. All leak detection drains permit free gravity drainage to a local building drain sump.

#### **12.3.8.4 Condensate Filters and Demineralizers System**

The Condensate Purification System (CPS) shields dose rates in uncontrolled areas less than 6  $\mu\text{Sv/hr}$  (0.6 mrem/hr). The CPS removes condensate system corrosion products and impurities resulting from condenser tube leakage, some radioactive material, activated corrosion products, and fission products carried over from the reactor.

#### **12.3.9 Radwaste Building**

Liquid and solid radioactive wastes generated during normal plant operations are transferred to the RWB for collection, holdup, and processing by Liquid Waste Management System (LWM) and the Solid Waste Management System (SWM).

1. LWM and SWM process subsystems include shielding and controls limiting accessible general area radiation levels to less than 10 mSv/hr (1000 mrem/hr). Transient radiation levels during filter media or waste container transfer operations may exceed these levels, but remote operation limits the average worker radiation dose to less than 150  $\mu\text{Sv/hr}$  (15 mrem/hr).
2. Design features minimizing occupational exposure include:
  - a. Remote pipe and equipment flushing
  - b. Remote viewing and handling equipment
  - c. Centralized sampling station minimizing exposure time
  - d. LWM and SWM tanks vent to RWB Heating, Ventilation, and Air Conditioning (HVAC) for processing
  - e. Radiation monitors with alarms provided both inside and outside RWB building liquid or solid RW processing rooms
3. Processing systems (pumps, valves, tanks, and skid-mounted process systems) are remotely operated from a central RWB control panel.
4. Remote operation for routine radioactive waste management system functions is performed from local RW control panel. This includes remote actuated valves for controlling process flow between process stations and permanently installed equipment.
5. The RWB process systems area accommodates modular shield walls that limits access and reduces radiation levels from waste processing equipment.
6. Dry active waste sorting, processing, and packaging operations are also performed in the RWB or TB Annex. Portable radiation detectors, portable shielding, and remote handling tools reduces radiation levels and occupational exposure.
7. Routine maintenance uses skid-mounted process systems for RW processing, reducing maintenance requirements for permanently installed systems.

##### **12.3.9.1 Liquid Waste Management System**

The LWM collects liquid wastes from floor drains, and other sources within the facility. Radioactive liquids generated during operation are segregated, collected, stored, and processed in the LWM.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

1. Radioactive contaminants are removed, and the bulk of the liquid is purified and either returned to the Condensate Storage Tank (CST) or reprocessed. The BWRX-300 is designed as a zero liquid release plant under normal conditions.
2. LWM is divided into subsystems where liquid wastes are segregated and processed according to the impurity type and chemical content. Segregating liquid waste minimizes the total waste produced through efficient processing.
3. The calculated individual dose from the annual release of radioactive liquid effluents meets the DNNP action levels.
4. The liquid effluent release controls based upon the DNNP, action levels, and discharge limits are provided in the POSAR.
5. Liquid waste removed and concentrated in filter media, ion exchange resins, and other forms is further processed into the SWM.
6. Radioactive waste tanks effluents discharge near the tank room ventilation exhaust register. Baffle plates are used in tank vents preventing liquid and entrained solids splashing out. No hard connections to the ventilation system are used to avoid liquid or moisture propagation through the ventilation system damaging the HEPA filters.
7. The RW system has significant holdup capacity in waste collection tanks and sample tanks that allows for reprocessing and minimizes effluent releases to the environment.

#### **12.3.9.2 Solid Waste Management System**

1. Radioactive solid waste is segregated for wet and dry packaging minimize total waste stream and offsite shipment and storage.
2. SWM activities are controlled remotely at a local control panel. Remote controlled operations include movement of casks and liners, filter handling, resin transport, and movement or reconfiguration of RW processing skids.

#### **12.3.9.3 Offgas System**

1. The release of radioactive material into the atmosphere is minimized by the OGS. The OGS holds gas allowing for decay of the offgas stream that contain krypton, xenon, iodine, nitrogen, and oxygen isotopes.
2. The release of gaseous radioactive effluents to the environs maintains exposure in unrestricted areas ALARA according to action levels.
3. The gaseous effluent release controls based upon the DNNP action levels, action levels, and discharge limits are provided in the POSAR.
4. OGS equipment is selected, arranged, and shielded maintaining occupational exposure ALARA.
5. OGS contains a recombiner, condenser, and activated carbon beds.
6. OGS provides sufficient holdup until the required fraction of the radionuclides has decayed, and the daughter products are retained by the charcoal.

#### **12.3.10 Reactor Building Spent Fuel and Components**

The RB structural design provides flow resistance in contaminated areas that hold up radioactive releases from containment.

**Spent Fuel Storage Design Bases:** The fuel storage pool has capacity for five refueling cycles and a complete core offload as well as some highly radioactive waste (e.g., filters) assuring safety under normal and postulated accident conditions. These safety measures include:

- Periodic inspection and testing of components important to safety
- Radiation Protection shielding
- Containment, confinement, and filtering systems
- Residual heat removal capability that is reliable and testable
- Ensuring fuel storage coolant inventory under accident conditions

The design allows for:

- Controlling airborne release
- Preventing coolant inventory drainage below the acceptable depth through the use of drains, gates, and weirs
- Coolant flow to spent fuel racks
- Fuel pool liner leakage detection and confinement

#### **12.3.10.1 Reactor Water Cleanup System**

1. CUW system that is in contact with the reactor coolant system is constructed of corrosion-resistant materials. For system piping, smooth bends are used instead of welded fittings. At certain locations within the system, stainless steel pipes with electro-polished finish are used to reduce corrosion product buildup.
2. CUW components include an RHX, a pressure reduction station, and associated valves located inside the TB.
3. The shielded cubicle CUW valves and piping reduces radiation levels, but entry is not allowed during normal operation. Most CUW valves and piping are in a shielded valve gallery adjacent to the cubicle and are remotely operated.
4. Chemical cleaning and decontamination connections use condensate to flush piping and equipment for decontamination prior to RHX maintenance pressure reduction station, and associated valves.
5. HVAC system limits the spread of contaminants from these shielded CUW cubicles by maintaining negative pressure.
6. Personnel access to the CUW cubicles for component maintenance is controlled to minimize personnel exposure.

#### **12.3.10.2 Control Rod Drive System**

The FMCRD design allows removal of the motor unit, position indicator probe, separation indicator probe and lower component for maintenance during plant outages without disturbing the upper assembly of the drive.

#### **12.3.10.3 Containment**

1. Containment and other safety features limit accident radiological effects and release are within acceptable limits.

2. Containment air that is inerted with nitrogen during normal operation is purged (specifically purging N-16) before access is allowed during shutdown conditions. Containment access is not permitted during normal operation above 25% power.
3. Prior to removing the reactor vessel head, airborne radioactive material sources are removed by either the containment purge exhaust system or the main condenser.

#### **12.3.10.4 Containment Leak Detection and Isolation System**

For process lines that penetrate the containment, leakage detection and isolation is provided. The leakage detection instrumentation initiates alarms at established limits and isolates the affected system(s).

#### **12.3.10.5 Plant Cooling Water System**

Plant cooling water is supplied to for the following fire protection measures:

1. Fire suppression devices are provided for activated carbon beds and radioactive filter that ignite due to high inlet temperatures.
2. Fire protection barriers and fire detection protects decontamination areas and RW storage areas.
3. Fire protection hoses are in low radiation areas.

#### **12.3.10.6 Secondary Control Room**

1. SCR shielding design protects personnel from radiological conditions during normal operation. Shielding permits access and occupancy of the SCR following an event where the MCR becomes uninhabitable.
2. Habitability contingencies are provided for continued occupancy under accident, station blackout conditions, and severe accidents.

#### **12.3.11 Control Building**

1. The MCR shielding design is based upon protecting personnel from radiation resulting from the most limiting design basis accident. Shielding permits access and occupancy of the MCR to ensure plant personnel exposure following an accident does not exceed dose limits.
2. The MCR is shielded against normal sources of radiation. Habitability contingencies are provided for continued occupancy under design basis accident, station blackout conditions and Design Extension Conditions without core melt. Severe accidents require operators relocate to the secondary control room.
3. Safe occupancy of the MCR during abnormal conditions (AOOs, DBAs) is provided in the design.

#### **12.3.12 Decommissioning ALARA Considerations**

The following plant design features ensure the plant is operated and maintained ALARA. These features aid in facilitating ALARA exposures during decommissioning:

1. Shielding design protects workers during maintenance or repairs and decommissioning.
2. The RB, TB, RWB, and TB Annex design consists of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings for large equipment removal to facilitate decommissioning.

3. The RW process systems located in the RWB are skid-mounted and facilitate decommissioning and ease of access, by allowing truck access, and skid loading/unloading.

### **12.3.13 Ventilation**

#### **12.3.13.1 General HVAC System ALARA Design Controls**

1. The following design objectives apply to building ventilation systems:
  - a. Systems ensure releases to the environment are ALARA.
  - b. Radionuclide air concentrations in accessible areas to personnel for normal plant surveillance and maintenance is below 10% of the derived air concentrations during normal power operation.
  - c. HVAC room concentration setpoints are determined ensuring airborne releases are ALARA and comply with the action levels in the POSAR.
2. The plant stack and the major streams feeding the stack are monitored by the PRM so that action is taken avoiding releases in excess of regulatory limits.
3. During normal operation purging, the containment exhausts air through the HVAC HEPA filter unit prior to stack discharge.
4. During an isolation event, if offsite or backup power is not available, outside air is supplied to MCR by Emergency Filter Unit (EFU) operation.
5. The EFUs are in closed areas preventing the spread of radioactive contamination and providing adequate space for maintenance activities. Before charcoal removal, radioactivity is allowed to decay to minimal levels.
6. The control room EFU ventilation systems filter unit operates during accident conditions providing pressurization and outside air.
7. The shielding wall thickness between the HVAC filter cubicles is sized so that the dose contribution in any cubicle does not exceed 250  $\mu\text{Sv/hr}$  (25 mrem/hr) under normal operation, ensuring filter maintenance exposure ALARA.

#### **12.3.13.2 Heating, Ventilation and Cooling Systems**

1. HVAC system limits airborne contamination by providing airflow patterns from areas of low contamination to higher contaminated areas. This is accomplished by creating negative pressure areas in contaminated cubicles maintaining air flow into each cubicle from the corridor area. The higher contaminated areas exhausts flow through the HEPA filters prior to the plant stack.
2. HVAC service in contaminated areas circulates air through the contaminated areas of the buildings in a once-through fashion. Flow is directed from corridors (point of highest pressure) to the equipment alcove rooms, to the area rooms, to air ducts, and finally to the exhaust ductwork for processing through HEPA filters.
3. The ventilation system in the fuel pool and equipment areas sweeps air from the fuel pool surface, removing the major portion of potential airborne contamination at the source.
4. The passage of ventilation ducts through walls and structures ensures the shielding efficiency is not reduced. HVAC duct penetrations used for exhaust are routed above personnel head height to minimize streaming effects to worker exposure. HVAC duct penetrations are in line of sight between radiation sources and occupied areas.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

5. Major HVAC equipment (e.g., fans, coolers) is located in dedicated low radiation areas to minimize maintenance worker exposures.
6. Airflow from areas of lower contamination potential to areas of greater contamination potential are maintained using a negative pressure in the more contaminated area.
7. Buildings containing radioactive materials are equipped with radiological monitoring connections (sample ports) in the exhaust ductwork targeting radiological system leakages.
8. In areas where HVAC air emergency filtration is required, joints, corners, sharp bends, obstructions, and dampers where radioactive contamination accumulates are minimized.
9. Fume hoods or similar devices are installed in laboratories, sampling rooms, and component decontamination rooms.
10. HEPA and charcoal filters are installed in potentially contaminated exhaust pathways.
11. HVAC filters that are highly contaminated are shielded from one another and from personnel access areas to minimize exposure.
12. Isolation dampers are installed at fans, whenever flow needs segregation for repair and maintenance.
13. Remote actuation of redundant filter trains is provided for high-activity areas within the building.
14. HEPA filters are equipped with differential pressure detection and remote indication.
15. The MCR and SCR envelopes maintain overpressure.
16. HEPA filters are located close to the radioactive source.
17. Respiratory protection is provided in cases where high airborne contamination levels contaminants are not removed efficiently by the HVS.

**12.3.14 Monitoring of Individuals and Working Areas (Instrumentation for Radiation Levels and Airborne Radioactivity)**

The following Process Radiation and Environmental Monitoring System (PREMS) radiation monitoring subsystems are provided for monitoring area radiation and airborne radioactivity within the plant:

- Containment Monitoring Subsystem (CMon)
- Process Sampling Subsystem (PS)
- Process Radiation Monitoring Subsystem (PRM)
- Area Radiation Monitoring Subsystem (ARM)

Airborne radioactivity in effluent releases and ventilation air exhausts is continuously sampled and monitored by the PREMS for noble gases, air particulates and halogens. Airborne contamination is sampled and monitored at each stack, in the offgas releases, and in the ventilation exhaust from the RB, RWB and TB. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used to check for airborne radioactivity in work areas prior to entry where radiation levels have the potential to exceed the allowable limits.

#### **12.3.14.1 Containment Monitoring Subsystem**

The CMon provides instrumentation for monitoring the following conditions inside containment:

- Hydrogen and oxygen concentration
- Pressure
- Water level
- Temperature
- Relative humidity
- Area radiation
- Gross gamma radiation for fission products

Signals are provided to the appropriate Distributed Control and Information System according to safety category. The Distributed Control and Information System processes each signal for display in the MCR and SCR, actuation of audible and visual alarms, and initiation of response functions such as reactor scram, containment valve isolation, feedwater isolation and PAM.

Monitoring is provided during normal reactor operation, AOOs, and post-accident conditions.

#### **12.3.14.2 Process Sampling Subsystem**

PS collects liquid and gaseous samples for analysis. This analytical information is used to monitor plant and equipment performance. This process subsequently informs plant operations if changes to operating parameters are necessary in order to meet acceptance criteria. This subsystem functions during all plant operational modes. Continuously sampled flows are routed from select locations in the process streams to the sampling stations. The sample flows enter the sample stations where pressure, temperature, and flow adjustments are made as necessary. Grab sampling facilities and special monitoring are provided for select areas (e.g., liquid radioactive waste). Continuous samples are diverted to continuous monitoring equipment that transmits data to the plant computer. Alarms indicate off-normal conditions.

#### **12.3.14.3 Process Radiation Monitoring Subsystem**

Airborne radioactivity in effluent and ventilation air exhausts are continuously sampled and monitored for noble gases, air particulates, and halogens by the PRM. PRM samples and monitors airborne contamination at the stack, in OGS streams, and in the ventilation exhaust from the RB, RWB, and TB. Detection of high radiation levels are alarmed in the MCR. Sampling in the exhaust lines is continuously analyzed. PRM facilitates periodic sampling for analysis. In addition to this instrumentation, portable air samplers are used to check for airborne radioactivity in work areas prior to entry where radiation levels could exceed allowable limits. Detectors in the PRM subsystem provide measurement indication of airborne radiological conditions from occupied areas.

#### **12.3.14.4 Area Radiation Monitoring Subsystem**

The ARM continuously measures, indicates and records gamma radiation levels at strategic locations throughout the plant except within containment. Alarms are activated in the MCR and SCR, as well as in local areas to warn personnel to take appropriate protective measures.

ARM detectors are located upstream of the CB and SCR AHUs. These detectors indicate and alarm in the MCR and SCR and are used by plant personnel in conjunction with portable monitors to evaluate airborne radiological conditions in occupied areas.

#### ***12.3.14.4.1 Area Radiation Monitoring Subsystems Description***

Every ARM channel consists of a gamma-sensitive detector and a digital area radiation processor. All channels are provided with local visual and audible alarms and local readouts. Additional readouts and alarms are provided by local auxiliary units. The output signals from the detectors are digitized and multiplexed for transmission to digital radiation monitors for measurement and display. The radiation signals are transmitted to the process computer for recording. Each radiation monitoring channel has adjustable trip alarm circuits, one for high radiation and the other for downscale indication (loss of sensor input). Also, each area radiation monitor has a built-in self-check for gross failures and activates an alarm on a power failure or an inoperative monitor. Auxiliary units with local audible alarms are provided in selected local areas that immediately warn occupants. Each area radiation monitor is powered from an uninterruptible power source that is continuously available during loss of offsite power.

#### ***12.3.14.4.2 Area Radiation Monitoring Subsystems Detector Locations and Sensitivity***

The appropriate locations and monitoring ranges of each area radiation monitor is finalized in detailed design phase.

#### ***12.3.14.4.3 Area Radiation Monitoring Subsystems Design Parameters and Requirements***

Two high-range radiation channels are provided on the refueling floor and fuel pool area to monitor radiation from a design extension condition Fuel Handling Accident. The conservative Out of Core Criticality analysis provided in Subsection 15.5.9.3 demonstrates that a criticality alarm system is not necessary at the BWRX-300 DNNP. An Out of Core Criticality accident in the refueling area is practically eliminated due to physical impossibility and supporting operating procedures. Practical elimination of this event is consistent with BWRX-300 Safety Strategy, described in Chapter 3, Subsection 3.1.8.

Defense Line 1 design provisions and supporting operating procedures provide design and administrative controls that effectively manage factors that influence system reactivity and the criticality. Design factors that influence reactivity are enrichment, mass, moderation, geometry, reflection, interaction and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical at any one time and under the most adverse moderation conditions feasible by un-borated water. The amount of water moderator available for internal flooding is limited by the storage location structure design making any credible possibility of an Out of Core Criticality physically impossible. As a result, the need for a criticality alarm system is practically eliminated consistent with BWRX-300 Safety Strategy as described in Chapter 3, Subsection 3.1.8.

Both process radiation monitors and area radiation monitors are in the fuel storage and associated handling areas.

Process radiation monitor alarms and isolation are provided for fuel pool area ventilation. Area radiation monitors with visual and audible alarms is also provided.

Final alarm setpoints established in the field are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures. The radiation alarm setpoint for each channel is set slightly above the background radiation level for the monitor location.



**12.3.15 References**

- 12.3-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.3-2 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 12.3-3 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.
- 12.3-4 USNRC Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants."
- 12.3-5 USNRC Regulatory Guide 1.54, "Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants."

## **12.4 Shielding**

The permanent shielding analysis is completed during detailed design phase. The evaluation identifies those areas experiencing elevated dose rates during power and shutdown conditions and are not mitigated by structure or component design or location. The anticipated dose rates and frequency of access are included in analysis. Using these results, system interaction and structural/component loading analyses, permanent shielding is installed in target areas/components. Permanent shielding is preferred to temporary shielding.

### **12.4.1 General Design Guides**

The primary objective of radiation shielding is protecting operating personnel, and the general public from radiation emanating from the reactor, the power conversion systems, the RW process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance in accordance with REGDOC-2.7.1, Sections 4.1 and 5.3 (Reference 12.4-1). Radiation shielding keeps radiation doses to equipment below levels where disabling radiation damage occurs.

The shielding requirements in the plant perform the following functions:

1. Limit, plant personnel, contractors and visitors, exposure ALARA and within guidance of ICRP Publication 103 (Reference 12.4-2) and CNSC SOR/2000-203 (Reference 12.4-3) limits
2. Limit personnel radiation exposure in the unlikely event of an accident to levels that are ALARA, conforming to the limits specified in REGDOC-2.5.2, Section 4.2.1 (Reference 12.4-4 and REGDOC-2.4.1, Section 4.3.2, (Reference 12.4-5) and ensuring the plant is maintained in a safe condition during and after an accident
3. Limit critical components radiation exposure within specified radiation tolerances assuring component performance and design life are not impaired

### **12.4.2 Design Description**

The following guidance is used in the BWRX-300 shielding design:

1. All systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone maps described in 12.6 indicate design radiation levels where equipment shielding contributes to achieving the dose rate in the area.
2. The source terms used in the shielding calculations are analyzed conservatively. Transient conditions, shutdown, and normal operating conditions are used ensuring a conservative source is used in the analysis. Shielding is based on design basis fission product concentrations in the coolant in addition to activation products. For components where N-16 is the major radiation source, a concentration based upon BWR fleet operating plant data is used.
3. Processing equipment is located as far as possible from personnel dwelling and passage areas minimizing shielding. Shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.
4. Penetrations through shield walls are shielded reducing radiation streaming through the penetrations. The approaches used to locate and shield penetrations are discussed in Subsection 12.3.6.
5. Wherever possible, radioactive piping is run in a manner minimizing plant personnel radiation exposure that includes:

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

- a. Minimizing radioactive pipe routing in corridors
  - b. Avoiding routing high-activity pipes through low radiation zones
  - c. Using shielded pipe trenches and pipe chases where routing of high-activity pipes in low-level areas cannot be avoided
  - d. Separating radioactive and non-radioactive pipes for maintenance
6. Maintaining acceptable levels at valve stations by using motor-operated or diaphragm valves. Minimizing worker radiation exposure by providing draining and flushing provisions in the valve design. Operator shielding is provided for manual valve applications using shield walls or valve stem extensions.
  7. Shielding is provided to permit access and occupancy of the MCR ensuring plant personnel exposure following an accident does not exceed the values set forth in REGDOC-2.5.2, Section 4.2.1 and REGDOC-2.4.1, Section 4.3.2 and supports compliance with REGDOC-2.5.2, Section 8.10.1.
  8. Provisions are made for shielding major radiation sources during ISI to reduce personnel exposure. Steel platforms are provided for ISI of the RPV nozzle welds and associated piping.
  9. The primary material used for shielding is concrete. Where special circumstances dictate, steel, lead, water, a modified high-density silicone elastomer, or a boron-laced refractory material is used.

#### **12.4.3 Shielding Method Design**

Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a Loss-of-Coolant Accident (LOCA) or a Fuel Handling Accident, are considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associated equipment and shielding are established to ensure conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity for the actual physical situation are incorporated.

#### **12.4.4 References**

- 12.4-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.4-2 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.
- 12.4-3 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 12.4-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 12.4-5 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."

## **12.5 Dose Constraints and Dose Assessment**

This section describes the Radiation Protection philosophy, objectives, and requirements used throughout the plant design. Radiation Protection ensures:

1. Compliance with regulatory dose limits
2. Radiation exposure during normal plant operation is ALARA with social and economic factors taken into account
3. Cumulative plant worker doses average less than one person-Sv/year over the operating life of a single unit plant

### **12.5.1 Interfaces for Protecting People and the Environment**

#### **12.5.1.1 Radiation Protection Objectives**

During normal operation and AOOs, the Radiation Protection Program objectives are:

1. Radiation exposures are kept within regulatory limits
2. Radiation exposures are kept ALARA with social and economic factors taken into account

Adherence to the first objective is demonstrated by comparing the calculated dose consequences with the dose limits specified in SOR/2000-203 (Reference 12.5-1). The second objective of keeping exposures ALARA, is a cost/benefit design decision for reducing dose exposures. This is accomplished by following the:

SOR/2000-203 (Reference 12.5-1) and REGDOC-2.7.1 (Reference 12.5-2).

#### **12.5.1.2 Normal Operation Dose Limits and Action Levels**

SOR/2000-203 (Reference 12.5-1) states in part: "every licensee shall implement a Radiation Protection Program and shall, as part of that program, (a) keep the amount of exposure to radon progeny and the effective dose and the equivalent dose received by and committed to persons as low as is reasonably achievable, social, and economic factors being taken into account."

#### **Nuclear Energy Worker**

The CNSC effective dose limits for a nuclear energy worker is an average of 20 mSv effective dose per year over a five-year period (100 mSv over five consecutive years), with no single year exceeding 50 mSv effective dose.

#### **Members of the Public**

Per CNSC Radiation Protection Regulations, the effective dose limit for a member of the public is 1 mSv per year from all sources of radiation other than natural background and medical exposures.

#### **Action Levels**

Action levels for the Radiation Protection Program are developed considering the guidance from REGDOC-2.7.1 (Reference 12.5-2). Typical action levels include external or internal doses above the planned ambient external or airborne hazards that are greater than anticipated in the design, and contamination found at levels significantly greater than expected with the plant or released to the public domain. These action levels are developed and available as part of the licence to operate.

#### **12.5.1.3 Limiting Exposures**

The BWRX-300 design minimizes radiation exposure and is consistent with CNSC REGDOC-2.7.1 (Reference 12.5-2).

#### **12.5.1.4 Waste Management**

Management of radioactive solid, liquid, and gaseous waste produced annually and during the operating life of the station is minimized according to ALARA principles and practices. Radioactive waste streams are controlled to limit worker and public doses and comply with regulatory limits. See Chapter 11 for a description of RW management systems.

The BWRX-300 design ensures public exposure from all radionuclides in effluent streams is less than 1 mSv/year (100 mrem/year). The following design and operational practices ensure the plant operates within the action levels:

1. Controlled and monitored liquid and gaseous waste releases
2. Liquid and gaseous waste activity is measured and recorded prior to release
3. The waste management systems have a target dose of 50  $\mu$ Sv per year averaged over the plant lifetime at the exclusion area boundary (see REGDOC-2.7.1) (Reference 12.5-2)

#### **12.5.1.5 Radiation Monitoring**

Radiation monitoring includes:

1. Detecting failed fuel in the core
2. Measuring external radiation in plant locations where plant workers and equipment are present
3. Monitoring, tracking, and recording personnel dosimetry
4. Monitoring, alarming, and isolating airborne contaminants
5. Providing offsite monitoring for radioactive material in the environment
6. Providing a radiation laboratory for analyzing environmental and biological samples, and workers monitoring
7. Providing a calibration facility for all plant radiation monitors

In the event of a release, the monitoring system provides:

1. Data to characterize radiological conditions around the plant and assess the need for offsite action
2. Data on radiological plant conditions facilitating safe evacuation, if necessary.

#### **12.5.1.6 Radiation Exposure Control Program**

The exposure risk for each radioactive plant system or component is estimated as part of a Radiation Exposure Control Program. The Radiation Exposure Control Program estimates the exposures involved in the operation, maintenance, inspection, repair, replacement, or decommissioning of potentially contaminated equipment.

Exposure considerations for plant workers include:

1. Breakdown of system work activities
2. Number of operators required for each activity
3. Time taken for each activity
4. External radiation field and internal airborne hazards

BWR operating experience and BWR operating data is used to inform decisions about estimated exposures.

**12.5.2 References**

12.5-1 Government of Canada SOR/2000-203, "Radiation Protection Regulations."

12.5-2 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

## **12.6 Radiation Zones and Access Requirements**

Radiation zones are established in all areas as a function of access requirements and area radiation sources. These zones are shown on Figures 12.6-1 through 12.6-17.

All areas of the plant have at least two assigned radiation zone classifications. The first classification is based on normal operation radiation fields and the second classification is based on the normal shutdown radiation fields. Additional classifications are assigned for selected off-normal events, as appropriate.

Updated radiation zones, or radiation zone maps are provided in the POSAR based on the general arrangement drawings, that show the normal operation and normal shutdown radiation zone classifications.

All radiation areas or enclosures have labels to show their radiation zone class. Access to radiation zones is controlled with locked doors in the case of high radiation areas (radiation levels greater than 1 mSv/h (100 mrem/hr)) with clearly marked entrances and chain barriers and marked entrances.

Technology infrastructure systems supporting permanent and temporary functions include:

- Physical access controls for radiologically controlled and significant areas
- Radiation work permit change stations
- Supplemental ARM with portable devices
- Equipment radiation monitoring
- Integrated security access and radiologically controlled area access
- Leak detection systems

### **12.6.1 Radiation Zones**

The following classifications for radiation zones are used:

#### **12.6.1.1 Class A Radiation Zones**

Class A radiation zones are non-radiological controlled zones and have free access and unlimited stay time. The areas outside the controlled area, including electrical equipment rooms, MCR, SCR and administrative areas, are Class A radiation zones. The radiation dose rate in these areas are allowed a maximum of 6  $\mu$ Sv/h (0.6 mrem/h).

#### **12.6.1.2 Class B Radiation Zones**

Class B radiation zones have restricted access and unlimited occupation time. This zone corresponds to the corridors, staircases, control panel areas, and areas around rooms with radioactive material. These areas do not have radioactive equipment or components. The shielded walls guarantee worker safety. The radiation dose rate in these areas reach 10  $\mu$ Sv/h (1 mrem/h), and there is no radioactive or environmental contamination. Access to these areas is through designated entry control points, and rooms with a higher radiation zone class are accessed from these control points.

#### **12.6.1.3 Class C Radiation Zones**

Class C radiation zones have restricted access and occupation time limited to 20 h/wk. These zones are areas of frequent access, such as sampling stations and valve operating rooms. The dose rate in contact with the existing equipment is not greater than 50  $\mu$ Sv/h (5 mrem/h), and there is a possibility of radioactive and/or environmental radiation.

#### **12.6.1.4 Class D Radiation Zones**

Class D radiation zones have restricted access and occupation time is limited to 4 h/wk. These areas are infrequently accessed and are rooms for valves or equipment where the dose rate on contact is no greater than 250  $\mu\text{Sv/h}$  (25 mrem/h). There is a possibility of radioactive and/or environmental radiation.

#### **12.6.1.5 Class E Radiation Zones**

Class E radiation zones have restricted access and occupation time is limited to 1 h/wk. These are occasionally accessed for equipment actuation (pumps, valves, heat exchangers and tanks). The contact dose rate with equipment is no greater than 1000  $\mu\text{Sv/h}$  (100 mrem/h), but the radiation level may be reduced without significantly altering the plant operating mode. There is a possibility of radioactive and/or environmental radiation.

#### **12.6.1.6 Class F to J Radiation Zones**

Class F to J radiation zones are high radiation zones that are areas accessible to individuals where radiation levels external to the body could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in one hour at 30 centimeters from the radiation source, or 30 centimeters from any surface that the radiation penetrates.

Access to Class F and higher zones is not permitted without specific authorization. This corresponds to equipment rooms that do not require attention for actuation, inspection, and/or maintenance, such as tanks, filters, and pipes with radiation levels greater than 1 mSv/h (100 mrem/h). Their radiation level cannot be reduced without significantly altering plant operation.

Typical non-accessible Class F to J, High and Very High radiation areas include the reactor vessel area, containment during power operation, the areas of the RB with numerous or large unshielded containment penetrations, the areas of spent fuel transfer whenever it is present, the fuel pool, the steam pipe tunnel during power operation, the FPC systems and the resin wash water collection tanks.

The relationship between radiation zone designations and accessibility requirements is presented in Table 12.6-1. The dose rate applicable for a particular zone is based on OPEX and represents design dose rates in a particular zone. They are not interpreted as the expected dose rates that would apply in all portions of that zone, for all types of work within that zone, or all instances of zone entry. BWR plants have been operating for several decades, and OPEX with similar design basis shows that only a small fraction of the permissible dose is received from radiation sources controlled by equipment layout, or the structural shielding provided. The BWRX-300 radiation zone approach accomplishes the ALARA objectives.

Access to areas in the plant is controlled and regulated by the given area zone. Areas with dose rates where an individual could receive a dose in excess of 1 mSv (100 mrem) in a period of one hour are locked and posted with "High Radiation Area" signs. Areas where an individual could receive a dose in excess of 5 Sv (500 rem) within a period of one hour at 1 meter from a radiation source, or 1 meter from any surface that the radiation penetrates are posted with "Very High Radiation Area" signs.



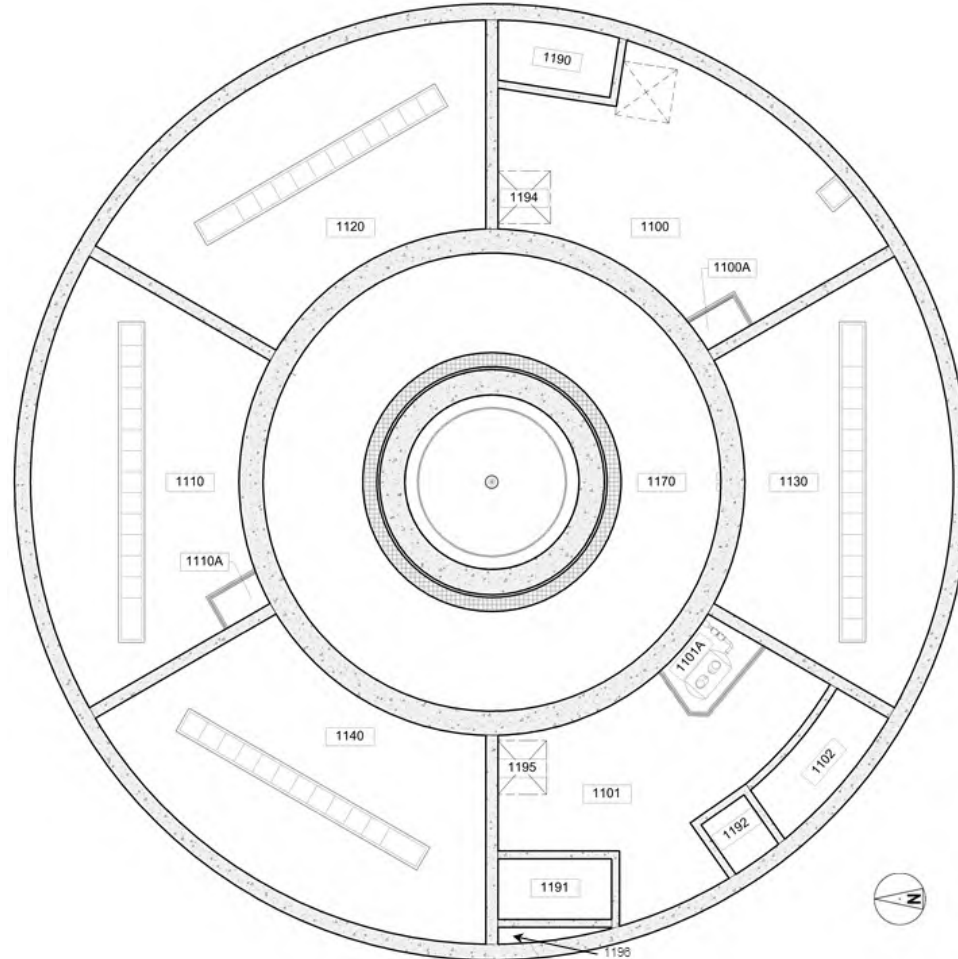
**Table 12.6-1: BWRX-300 Radiation Zone Classifications**

Zone	Dose Rate	Occupancy limit	Description
A	$\leq 6 \mu\text{Sv/h}$ (0.6 mrem/h)	40 hr/wk	Uncontrolled and unlimited access
B	$\leq 10 \mu\text{Sv/h}$ (1 mrem/h)	40 hr/wk	Controlled and unlimited access
C	$\leq 50 \mu\text{Sv/h}$ (5 mrem/h)	20 hr/wk	Controlled and limited access
D	$\leq 250 \mu\text{Sv/h}$ (25 mrem/h)	4 hr/wk	Controlled and limited access
E	$\leq 1 \text{ mSv/h}$ (100 mrem/h)	1 hr/wk	Controlled and limited access
F <sup>1</sup>	$\leq 10 \text{ mSv/h}$ (0.1 rem/h)	5 hr/year	Limited and controlled access with special authorization permit required
G <sup>1</sup>	$\leq 100 \text{ mSv/h}$ (10 rem/h)	30 min/year	(Same as Zone F above)
H <sup>1</sup>	$\leq 1 \text{ Sv/h}$ (100 rem/h)	No access	(Same as Zone F above)
I <sup>1</sup>	$\leq 5 \text{ Sv/h}$ (500 rem/h)	No access	(Same as Zone F above)
J <sup>2</sup>	$> 5 \text{ Sv/h}$ (500 rem/h)	No access	Limited and controlled access with special authorization permit required

(1) High radiation sources present. Limit and controlled access with special authorization permits required.

(2) Very High radiation sources present. Very limited and controlled access with special authorization permits required.

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radiation Zones Level -34.0**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1100	ENTRY -34.0	B	A
1101	SERVICE -34.0	B	A
1102	ERO STORAGE -34.0	A	A
1110	FMCRD GROUP 1 CONTROLS	D	C
1120	FMCRD GROUP 2 CONTROLS	C	B
1130	FMCRD GROUP 3 CONTROLS	C	B
1140	FMCRD GROUP 4 CONTROLS	C	B
1100A	SUMP 1	F	E
1101A	SUMP 2	D	C
1110A	SUMP 3	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELLWELL A	B	A
1191	STAIRWELLWELL B	B	A
1192	ELEVATOR	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

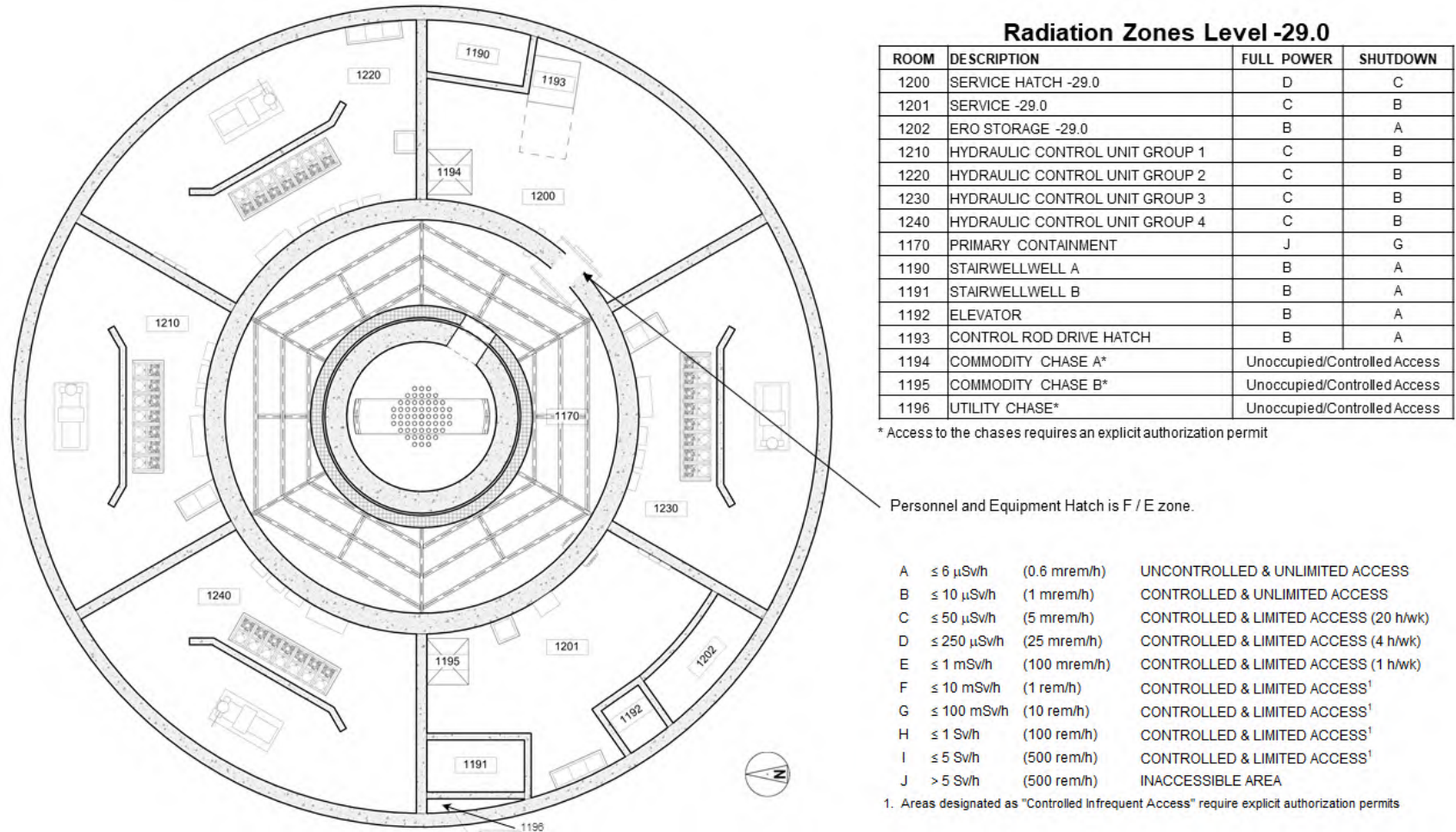
\* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

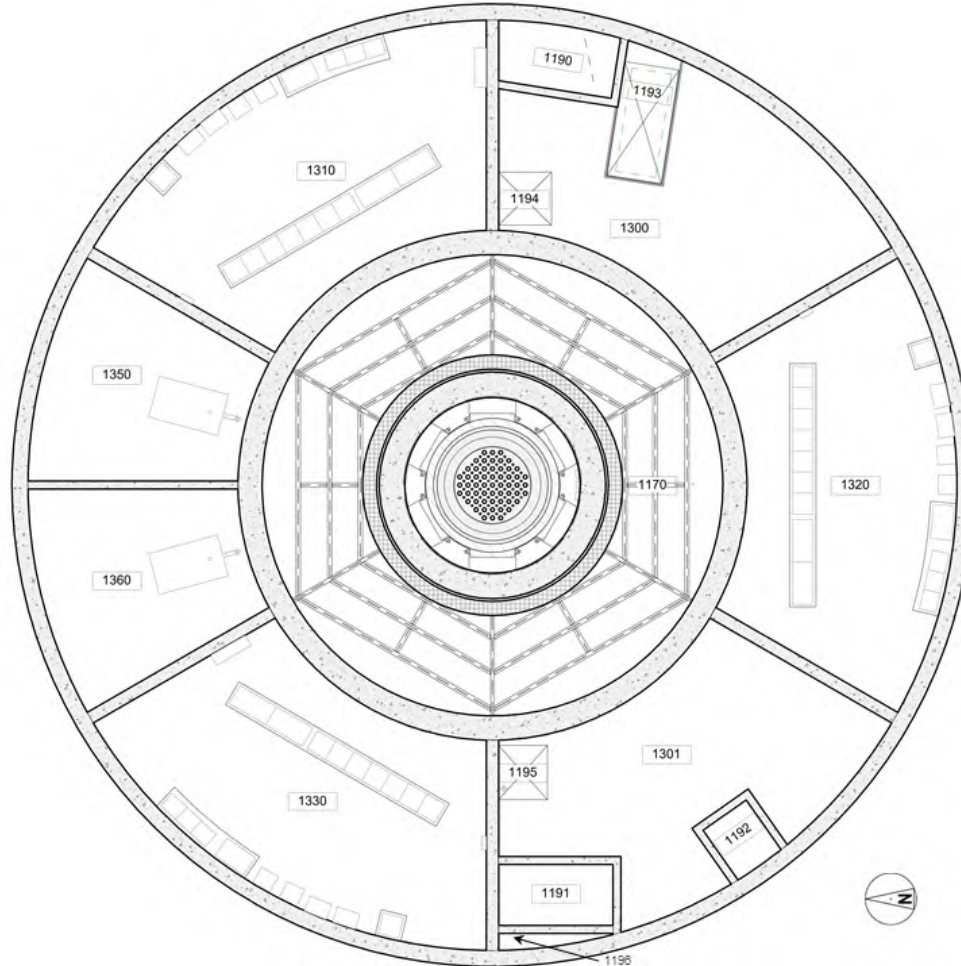
1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-1: Reactor Building Level -34.0 Meters Radiation Zones**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 12.6-2: Reactor Building Level -29.0 Meters Radiation Zones**



**Radiation Zones Level -21.0**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1300	ENTRY -21.0	B	A
1301	SERVICE -21.0	B	A
1310	DIV 1 DCIS & ELECTRICAL	B	A
1320	DIV 2 DCIS & ELECTRICAL	B	A
1330	DIV 3 DCIS & ELECTRICAL	B	A
1350	SHUTDOWN COOLING PUMP A	D	C
1360	SHUTDOWN COOLING PUMP B	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

\* Access to the chases requires an explicit authorization permit

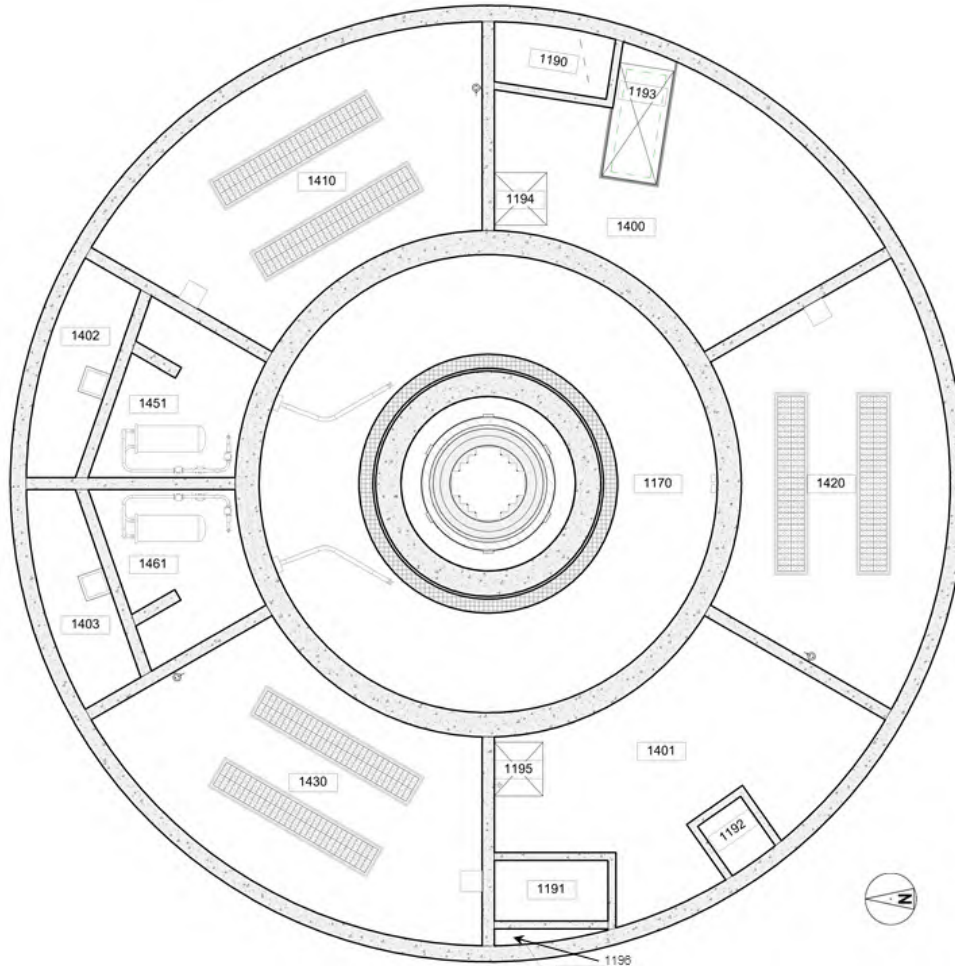
A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-3: Reactor Building Level -21.0 Meters Radiation Zones**



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radiation Zones Level -14.5**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1400	ENTRY -14.5	B	A
1401	SERVICE -14.5	B	A
1402	SHUTDOWN COOLING ENTRY A	B	A
1403	SHUTDOWN COOLING ENTRY B	B	A
1410	DIVISION 1 BATTERY ROOM	B	A
1420	DIVISION 2 BATTERY ROOM	B	A
1430	DIVISION 3 BATTERY ROOM	B	A
1451	SHUTDOWN COOLING A	D	C
1461	SHUTDOWN COOLING B	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR A	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

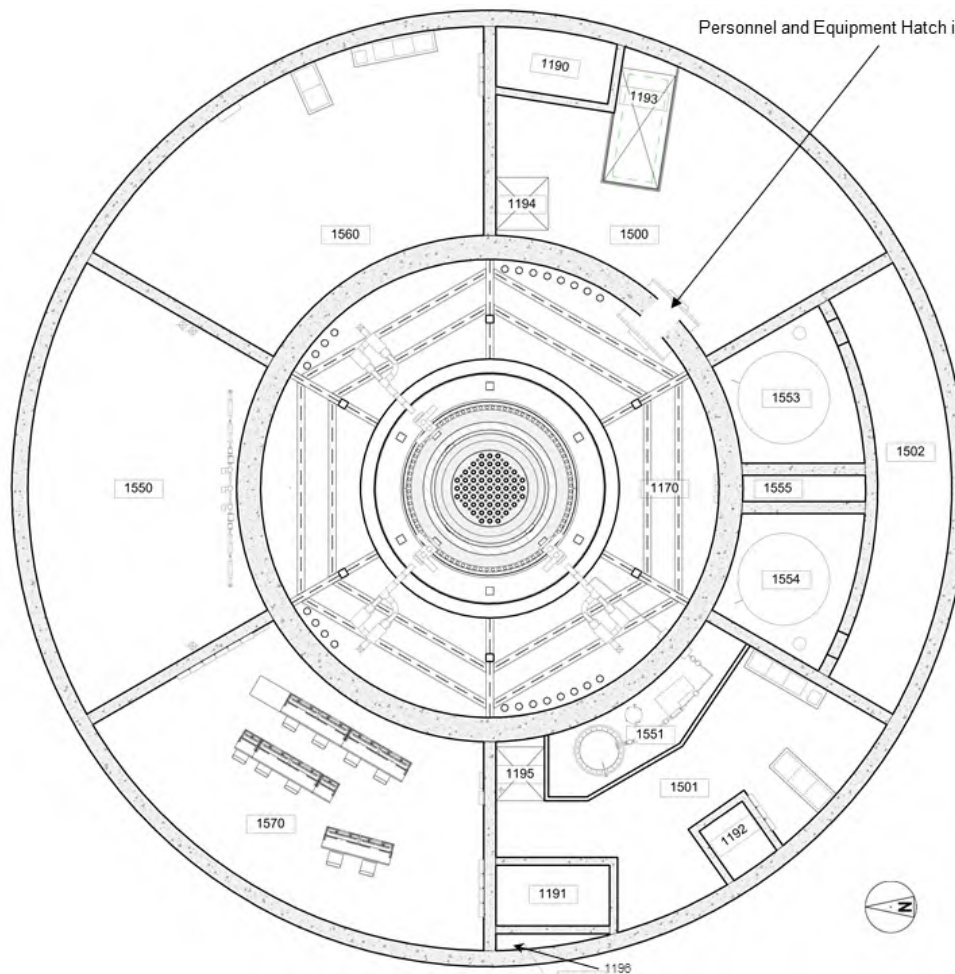
\* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-4: Reactor Building Level -14.5 Meters Radiation Zones**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radiation Zones Level -8.5**

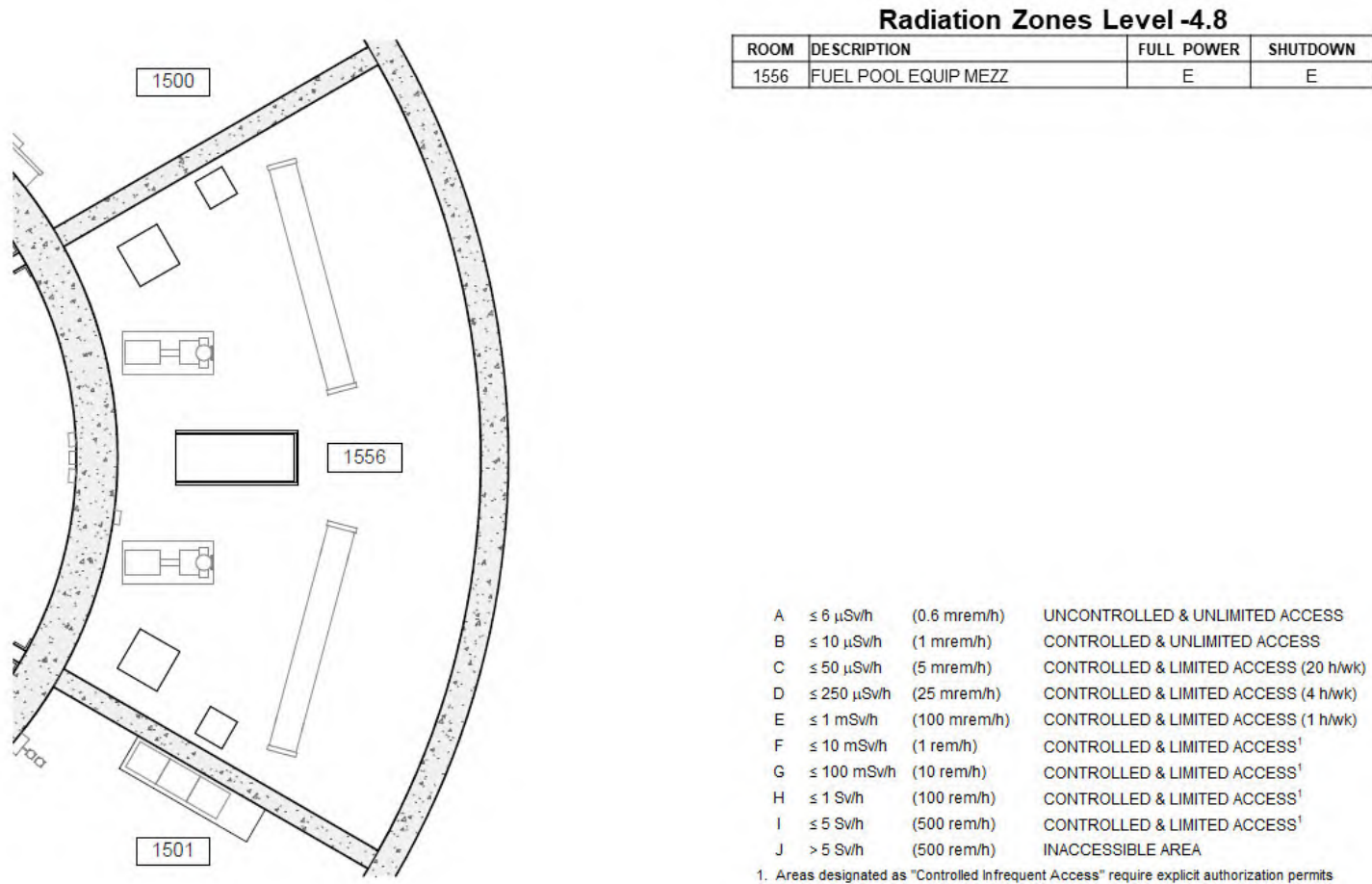
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1500	SERVICE HATCH -8.5	E	D
1501	SERVICE -8.5	B	A
1502	CORRIDOR -8.5	C	B
1550	SHUTDOWN COOLING PIPING	D	C
1551	BORON INJECTION	C	B
1553	FUEL POOL DEMIN A	C	B
1554	FUEL POOL DEMIN B	C	B
1555	FUEL POOL EQUIP STAIRWELL	C	B
1560	ELECTRICAL DISTRIBUTION	B	A
1570	SECONDARY CONTROL ROOM	B	A
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

\* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

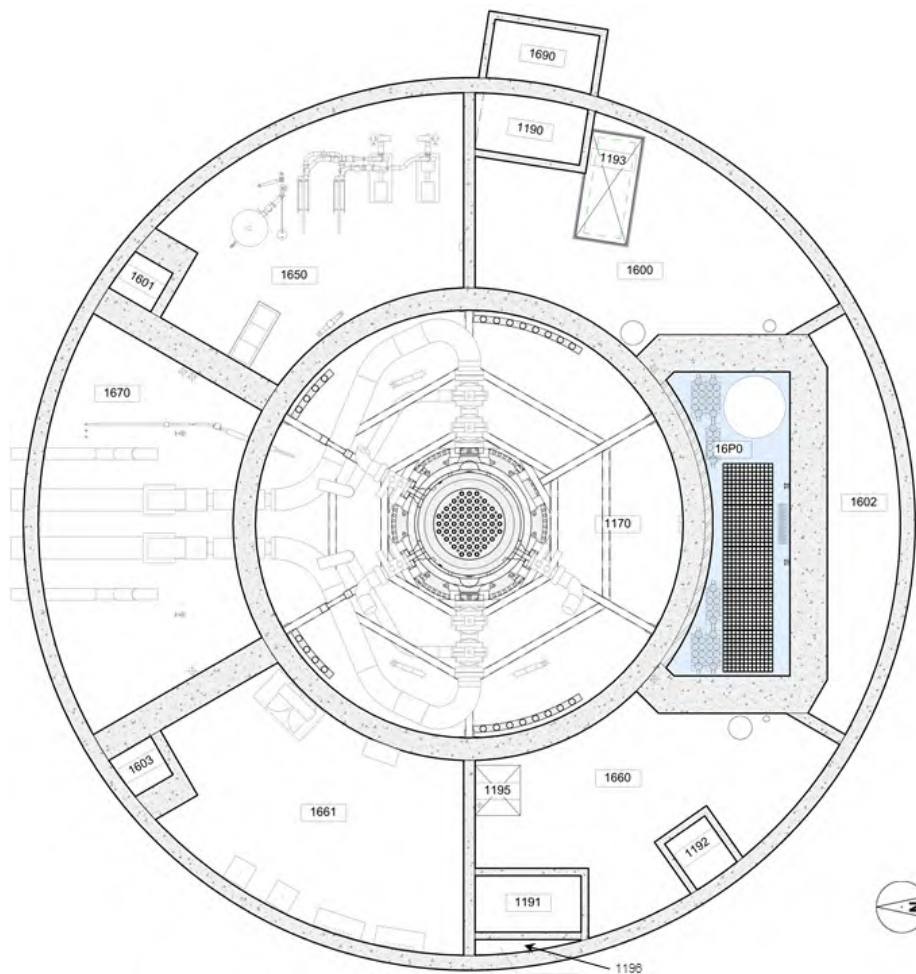
**Figure 12.6-5: Reactor Building Level -8.5 Meters Radiation Zones**



**Figure 12.6-6: Fuel Pool Mezzanine - -4.8 Meters Radiation Zone**



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radiation Zones Level 0**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	C	B
1601	LABYRINTH A	F	C
1602	HALLWAY	B	B
1603	LABYRINTH B	F	C
1650	ICC COOLING	D	C
1660	SERVICES 0 A	C	B
1661	SERVICES 0 B	D	C
1670	MS/FW PIPING	H	C
16P0	FUEL POOL	I	I
1690	STAIRWELL C	B	A
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR A	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1195	COMMODITY CHASE B*	Unoccupied/No Radiation Zone	
1196	UTILITY CHASE*	Unoccupied/No Radiation Zone	

\* Access to the chases requires an explicit authorization permit

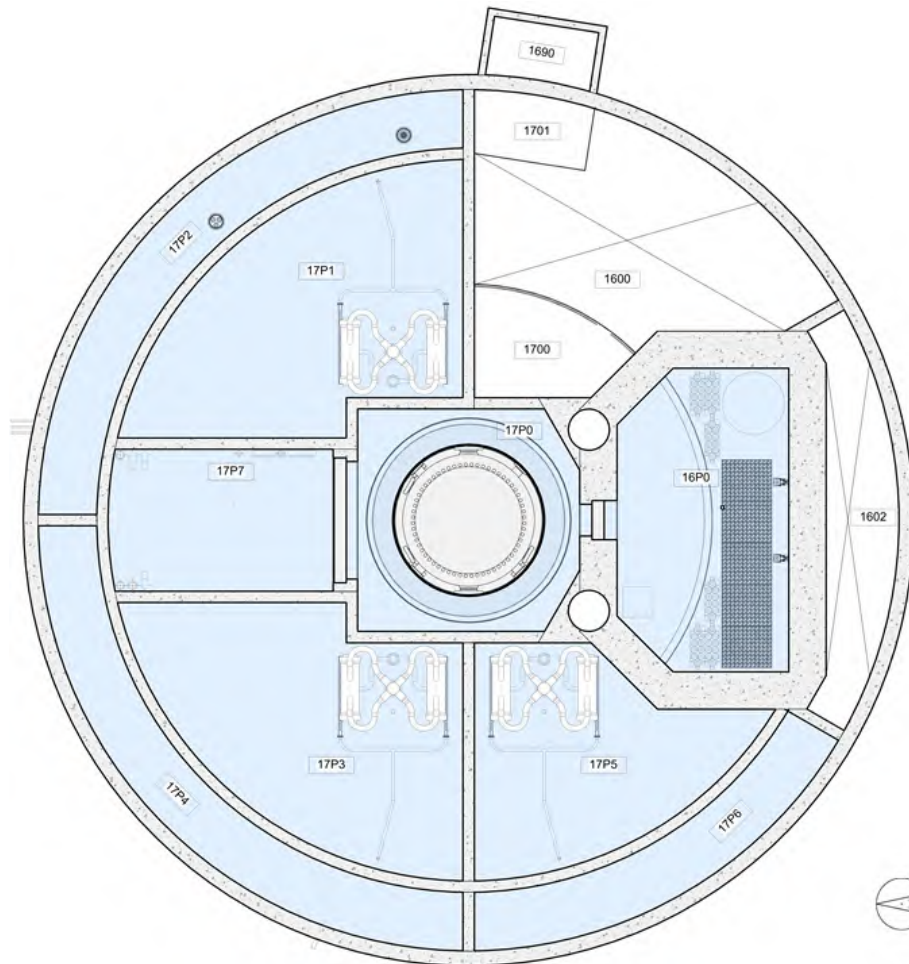
A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-7: Reactor Building Level 0.0 Meters Radiation Zones**



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radiation Zones Level 4.9**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	C	B
1602	HALLWAY	B	B
1700	TRUCK BAY MEZZANINE 1	B	A
1701	TRUCK BAY MEZZANINE 2	B	B
16P0	FUEL POOL**	I	I
17P0	REACTOR CAVITY POOL**	J	F
17P1	ISOLATION CONDENSER A	C	C
17P2	ISOLATION CONDENSER POOL A	B	B
17P3	ISOLATION CONDENSER B	C	C
17P4	ISOLATION CONDENSER POOL B	B	B
17P5	ISOLATION CONDENSER C	C	C
17P6	ISOLATION CONDENSER POOL C	B	B
17P7	EQUIPMENT POOL**	C	D
1690	STAIRWELL C	B	A

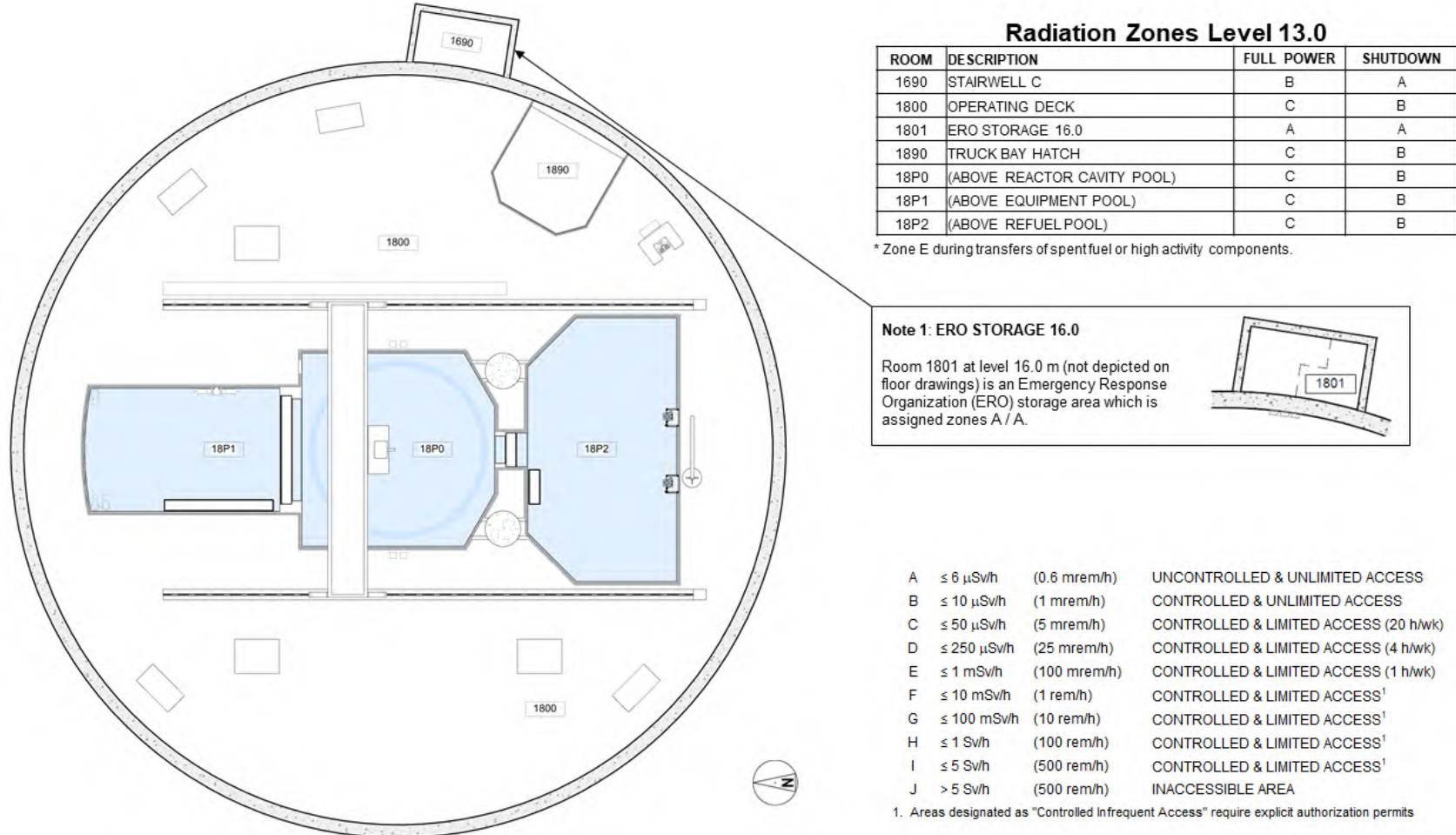
\*\* Zone J during spent fuel transfers

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-8: Reactor Building Level 4.9 Meters Radiation Zones**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



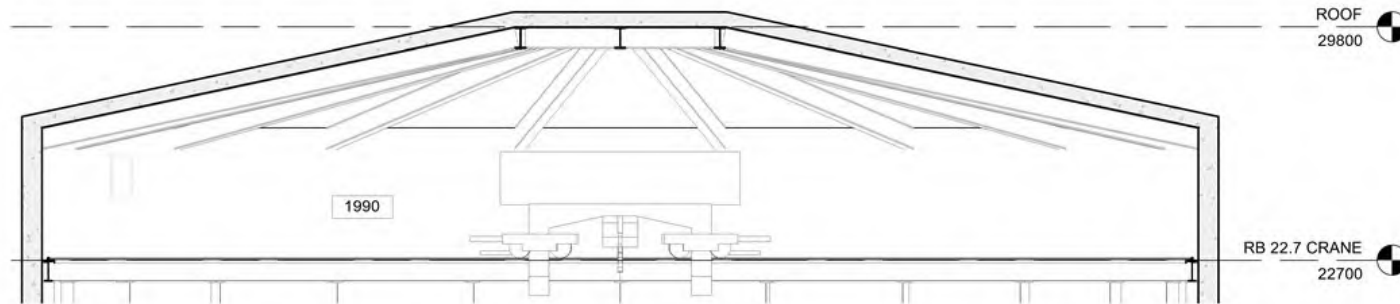
**Figure 12.6-9: Reactor Building Level 13.0 Meters Radiation Zones**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

**Reactor Building Radiation Zones Level 22.7**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1990	POLAR CRANE AREA*	C	B

\* Level 22.7 zone increases to D during transfers of spent fuel or high activity components.



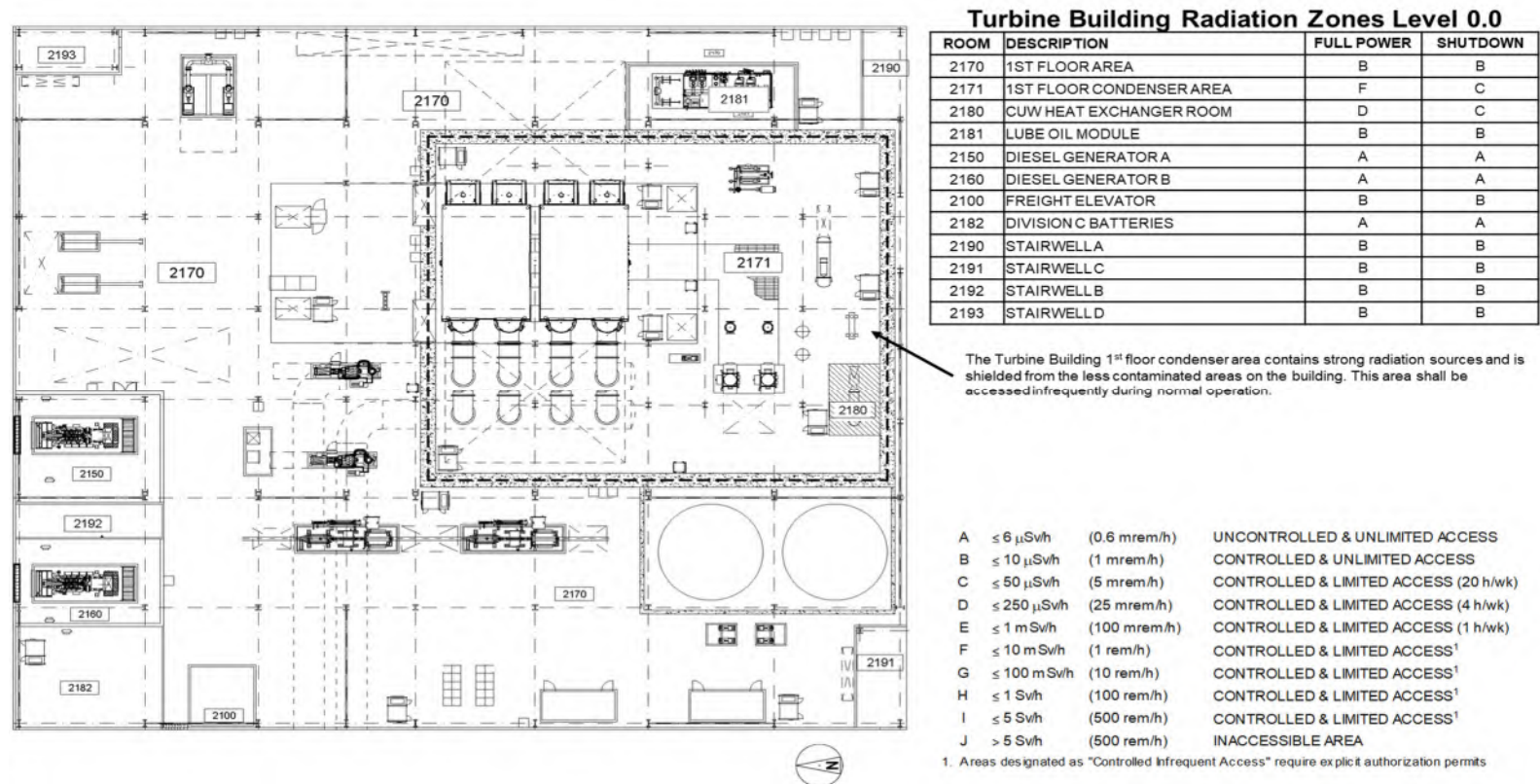
A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-10: Reactor Building Level 20.8 Meters Radiation Zones**

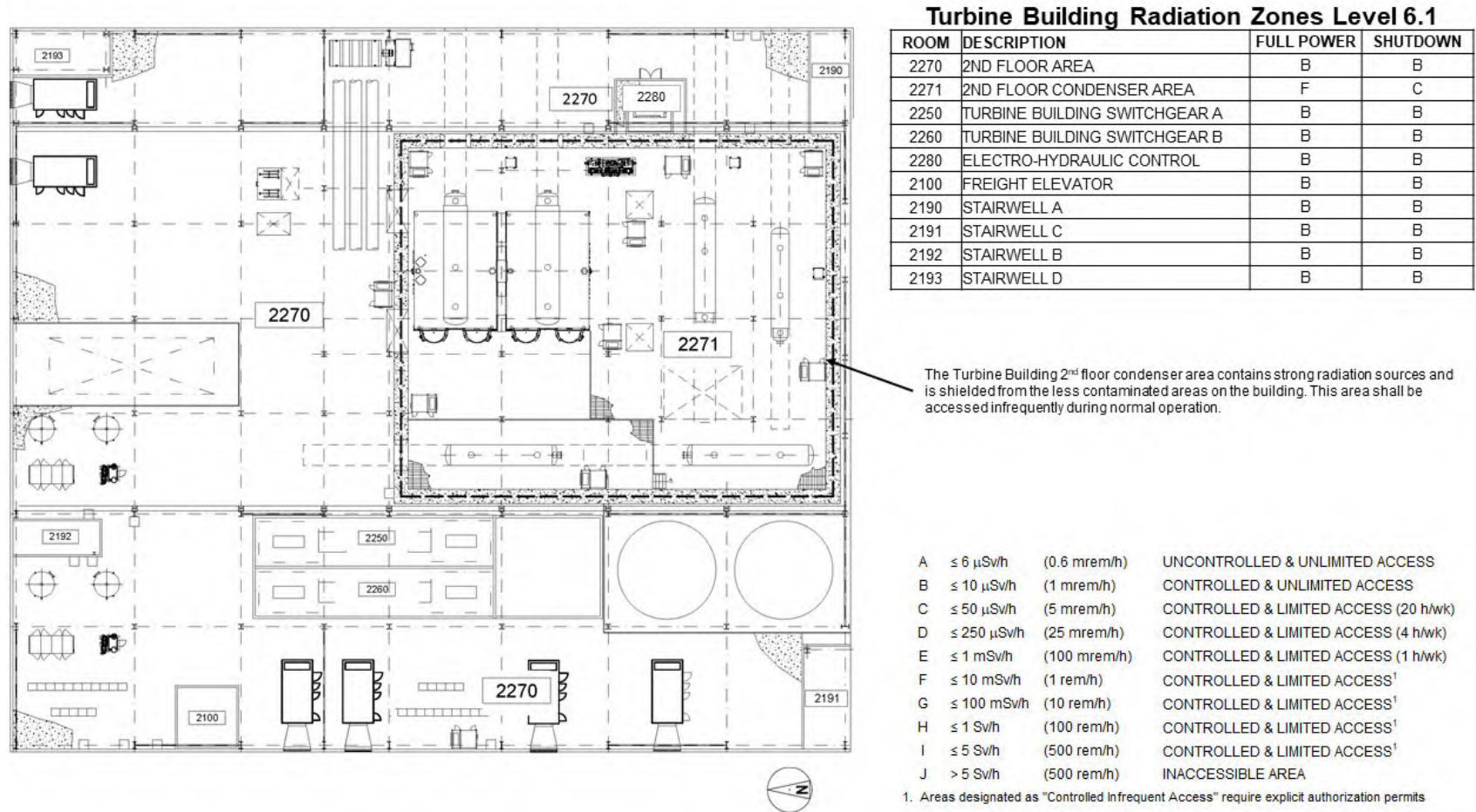


NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



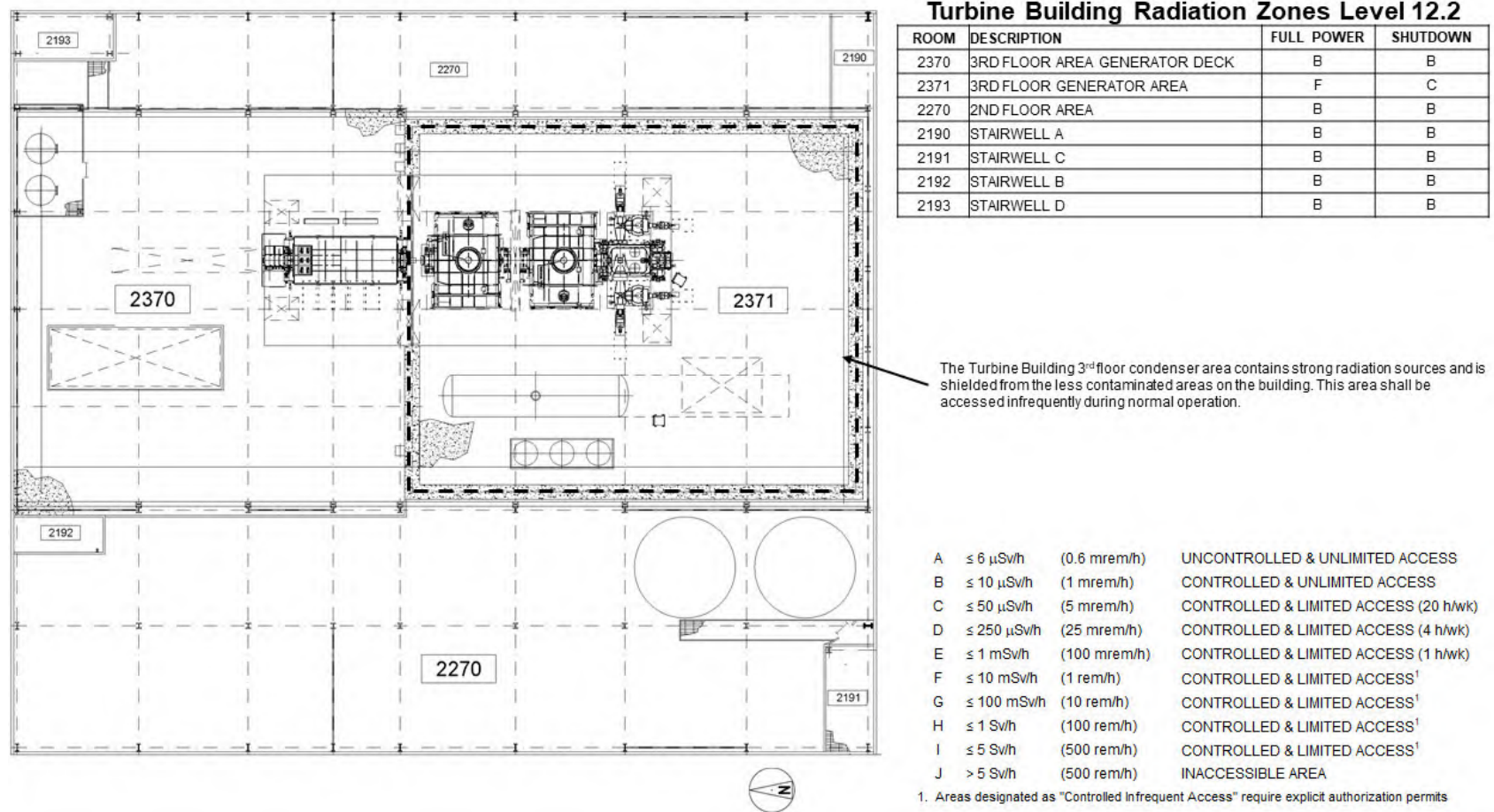
**Figure 12.6-11: Turbine Building Level 0.0 Meters Radiation Zone**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 12.6-12: Turbine Building Level 6.1 Meters Radiation Zone**

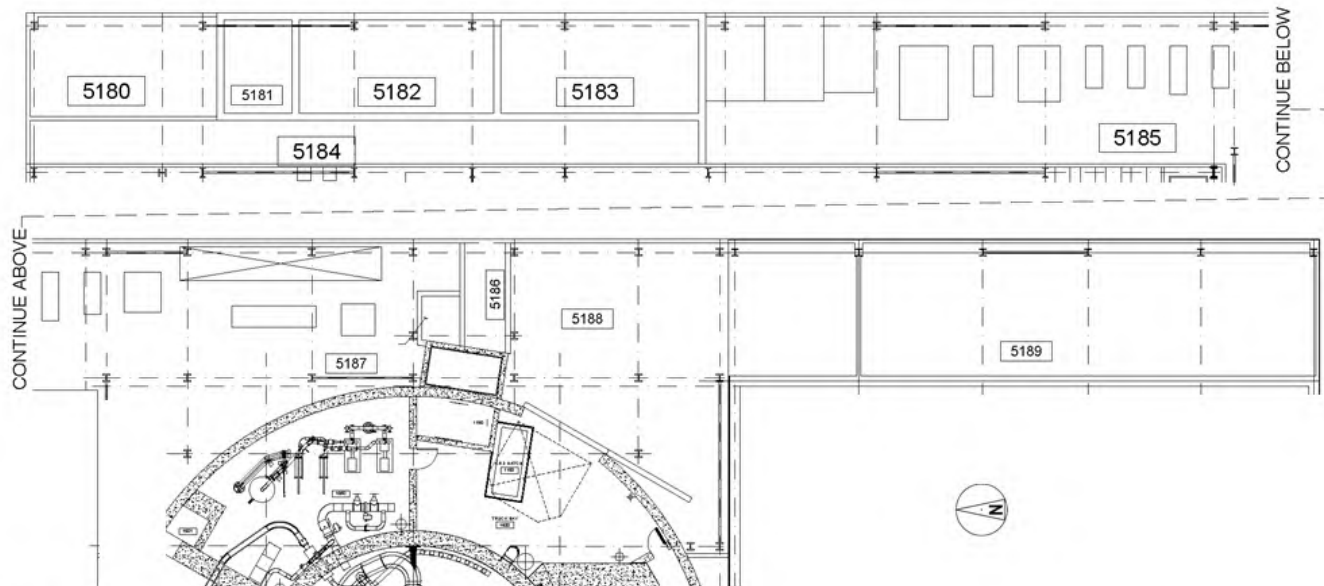
NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 12.6-13: Turbine Building Level 12.2 Meters Radiation Zone**



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Plant Services Building Radiation Zones Level 0.0**

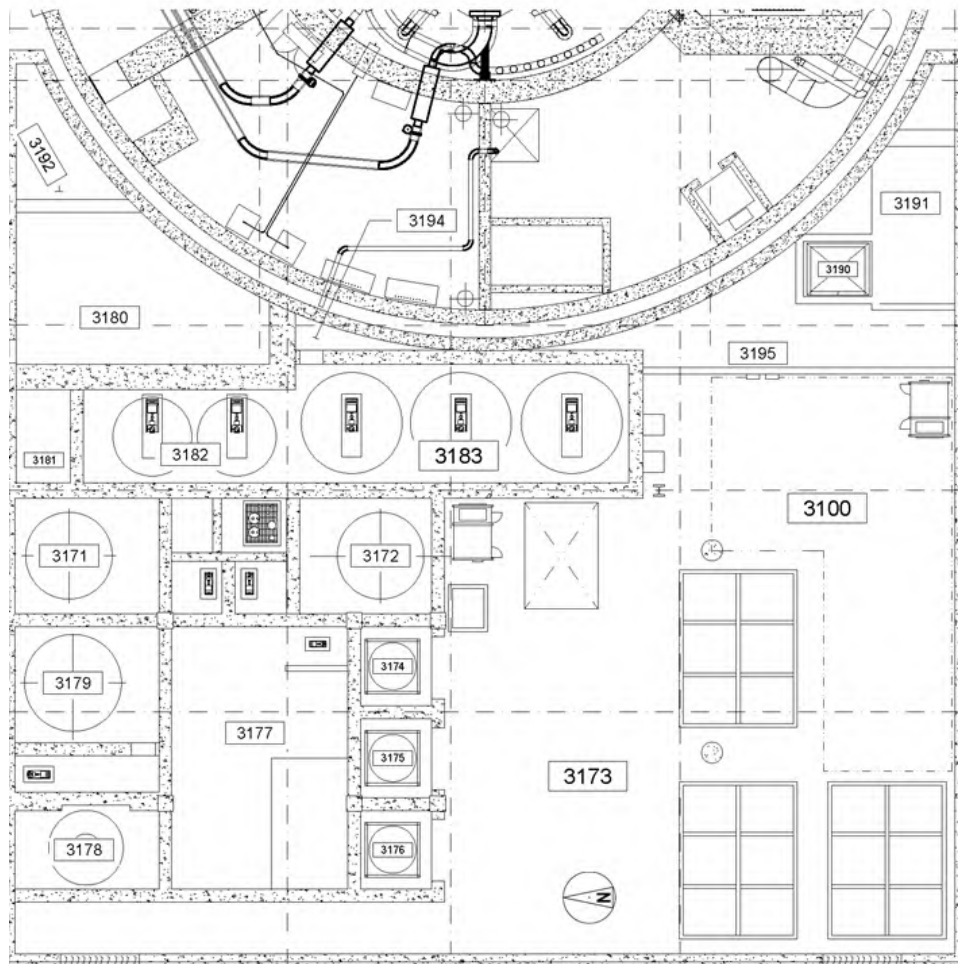
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5180	OFFICE SPACE	A	A
5181	I&C CALIBRATION AREA	B	B
5182	CONTAMINATED PART/TOOL STORAGE	C	C
5183	DECONTAMINATION AREA	D	D
5184	PLANT SERVICES BUILDING HALLWAY	B	B
5185	HOT MACHINE SHOP	D	D
5186	REACTOR BUILDING EGRESS PASSAGE	B	B
5187	AUX RCA ACCESS CONTROL ROOM	B	B
5188	TRUCK SPACE (CASK REMOVAL)	C	C
5189	STORAGE AREA (NEW FUEL, FMCRD, ETC.)	C	C

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-14: Plant Services Building 0.0 Meters Radiation Zone**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radwaste Building Radiation Zones Level 0.0**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
3100	DRESS OUT AREA / EME EQUIPMENT	A	A
3171	SLUGE 1 (K20)	H	H
3172	SLUGE 2 (K20)	H	H
3173	FILTERING SKID AREA	E	E
3174	HIGH INTEGRITY CONTAINER 1 (K20)	G	G
3175	HIGH INTEGRITY CONTAINER 2 (K20)	G	G
3176	HIGH INTEGRITY CONTAINER 3 (K20)	G	G
3177	DEWATERING PUMP ROOM	H	H
3178	DRUM EVAPORATOR (K20)	E	E
3179	SPENT RESIN (K20)	I	I
3180	OFFGAS ABSORBER VESSEL AREA	E	E
3181	SAMPLE (K10) TANKS ENTRY AREA	D	D
3182	SAMPLE (K10) TANKS	D	D
3183	COLLECTION (K10) TANKS	D	D
3190	ELEVATOR	B	B
3191	STAIRWELL A	B	B
3192	FLOOR AREA	B	B
3194	PIPE CHASE*	Unoccupied/Controlled Access	
3195	REACTOR BUILDING EGRESS PATHWAY	B	B

\* Access to the chases requires an explicit authorization permit

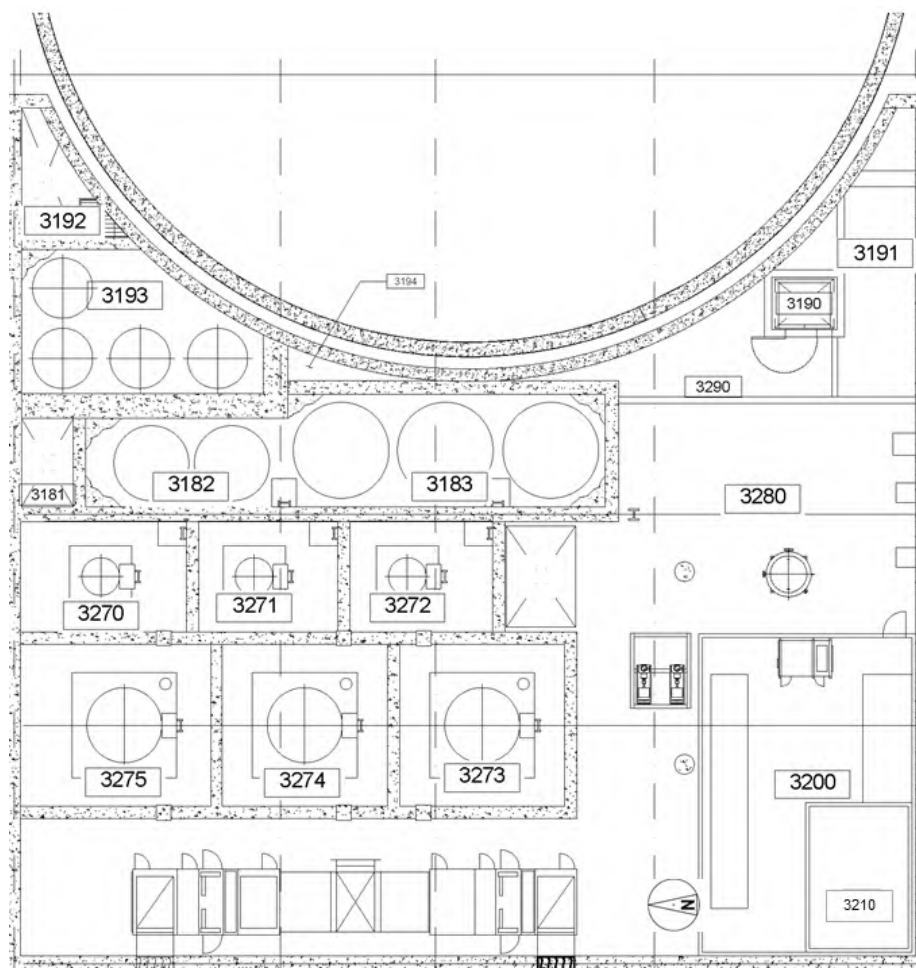
A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-15: Radwaste Building Level 0.0 Meters Radiation Zone**



NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radwaste Building Radiation Zones Level 6.1**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
3181	SAMPLE (K10) TANKS ENTRY AREA	D	D
3182	SAMPLE (K10) TANKS	D	D
3183	COLLECTION (K10) TANKS	D	D
3190	ELEVATOR	B	B
3191	STAIRWELL A	B	B
3192	FLOOR AREA	B	B
3193	UPPER OFFGAS ABSORBER VESSEL AREA	E	E
3194	PIPE CHASE*	Unoccupied/Controlled Access	
3200	LABORATORY	B	B
3210	COUNTING ROOM	B	B
3270	CONDENSATE DEMINERALIZER PRE-FILTER 1	E	E
3271	CONDENSATE DEMINERALIZER PRE-FILTER 2	E	E
3272	CONDENSATE DEMINERALIZER PRE-FILTER 3	E	E
3273	CONDENSATE POLISHER 3 (DEMINERALIZER)	D	D
3274	CONDENSATE POLISHER 2 (DEMINERALIZER)	D	D
3275	CONDENSATE POLISHER 1 (DEMINERALIZER)	D	D
3280	RADWASTE BUILDING 2ND FLOOR AREA	B	B
3290	REACTOR BUILDING EGRESS PATHWAY	B	B

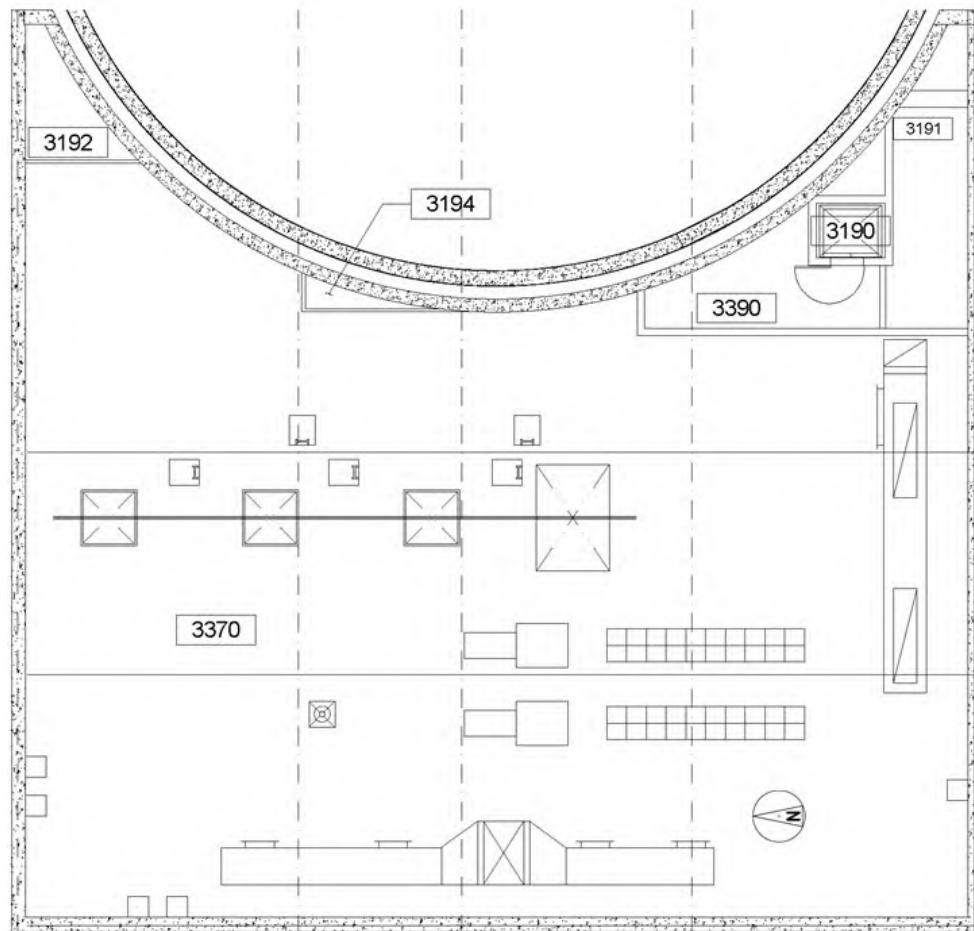
\* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-16: Radwaste Building Level 6.1 Meters Radiation Zone**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION



**Radwaste Building Radiation Zones Level 13.0**

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
3190	ELEVATOR	B	B
3191	STAIRWELL A	B	B
3192	FLOOR AREA	B	B
3194	PIPE CHASE*	Unoccupied/Controlled Access	
3370	CHILLERS EQUIPMENT/PIPING AREA	C	C
3390	REACTOR BUILDING EGRESS PATHWAY	B	B

\* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS <sup>1</sup>
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

**Figure 12.6-17: Radwaste Building Level 13.0 Meters Radiation Zone**

NEDO-33962 REVISION 0  
NON-PROPRIETARY INFORMATION

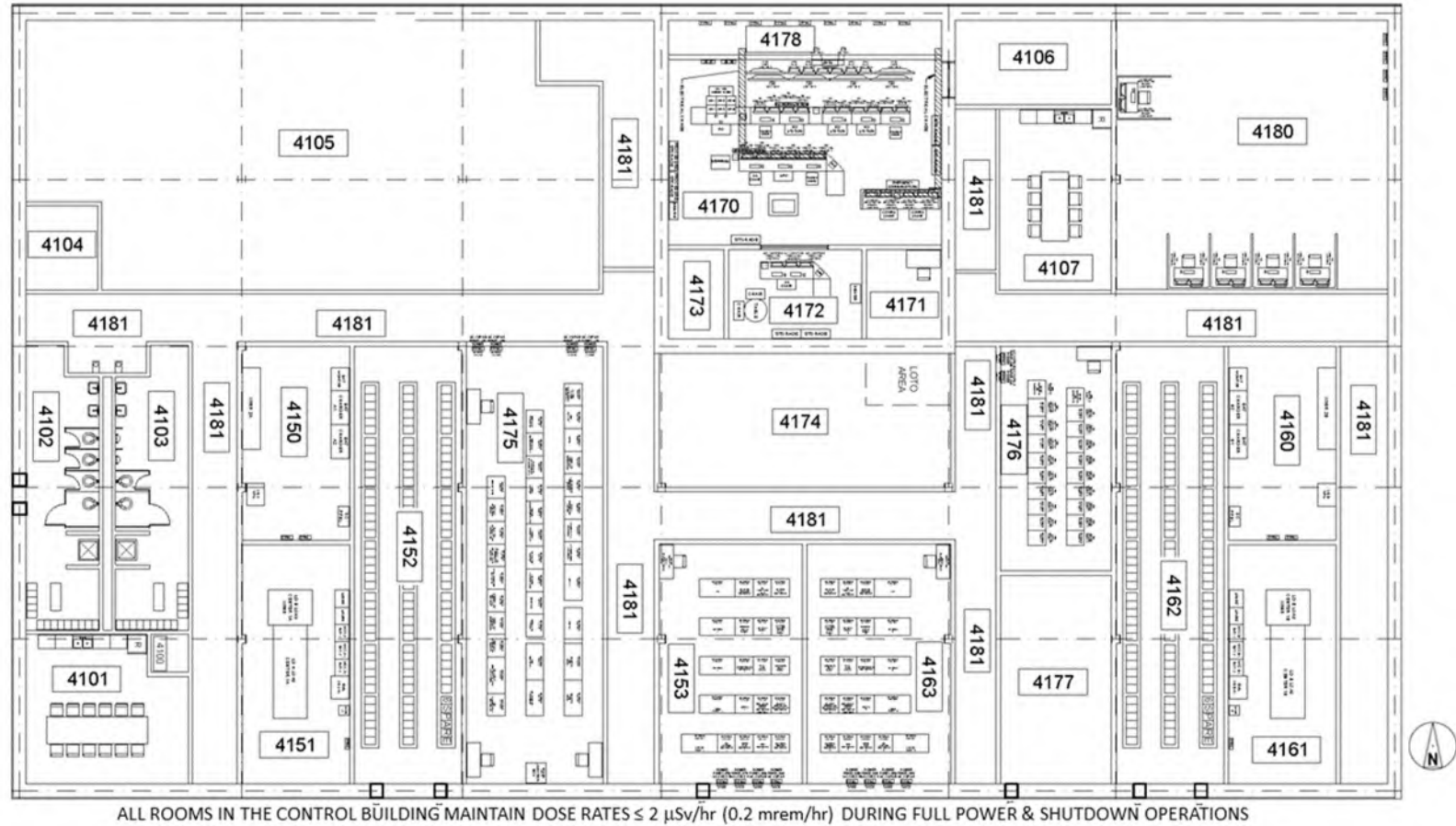


Figure 12.6-18: Control Building Level 0.0 Meters Radiation Zone

## **12.7 Radiation Protection Program**

The Radiation Protection Program is described in the Licence to Construct Application, Section 4.7, Radiation Protection, and is expected to be revised with the Licence to Operate application.

The Radiation Protection Program is consistent with the guidance set forth in CNSC REGDOC-2.7.1, "Radiation Protection" (Reference 12.7-1). The Radiation Protection Program consists of a series of procedures and programs that keep radiation exposure to workers and the public as low as reasonably achievable.

The Radiation Protection Program includes the following elements:

- Radiation protection organization, oversight, and administration
- Training and qualifications
- Process and procedures
- Radiation zones classification
- Program measures to limit exposure and dose
- Equipment and instrumentation
- Contamination control
- Emergency planning

### **12.7.1 References**

12.7-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33963

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 13  
Conduct of Operations**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AM	Aging Management
AOO	Anticipated Operational Occurrence
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
DBA	Design Basis Accident
DNNP	Darlington New Nuclear Project
EME	Emergency Mitigating Equipment
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
GEH	GE Hitachi Nuclear Energy
HFE	Human Factors Engineering
IAEA	International Atomic Energy Agency
OPEX	Operating Experience
OLC	Operational Limits and Conditions
OPG	Ontario Power Generation
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
SAMG	Severe Accident Management Guideline
SSC	Structures, Systems, and Components



## TABLE OF CONTENTS

<b>13.0 CONDUCT OF OPERATIONS .....</b>	<b>13-1</b>
13.1 Organizational Structure of Operating Organization.....	13-1
13.1.1 Organizational Structure .....	13-2
13.1.2 Qualifications of Plant Personnel .....	13-3
13.2 Training .....	13-4
13.2.1 General.....	13-4
13.2.2 Training Managed System Plan .....	13-5
13.3 Implementation of the Operational Safety Program .....	13-7
13.3.1 General Implementation.....	13-7
13.3.2 Conduct of Operations.....	13-7
13.3.3 Maintenance, Surveillance, Inspection and Testing .....	13-11
13.3.4 Core Management and Fuel Handling .....	13-12
13.3.5 Aging Management and Long-Term Operation .....	13-13
13.3.6 Control of Modifications .....	13-14
13.3.7 Program for the Feedback of Operating Experience .....	13-17
13.3.8 Documents and Records .....	13-17
13.3.9 Outages.....	13-18
13.4 Plant Procedures and Guidelines .....	13-20
13.4.1 Administrative Procedures.....	13-20
13.4.2 Operating Procedures.....	13-21
13.4.3 Procedures and Guidelines for Operating the Plant During Accidents..	13-22
13.5 Nuclear Safety and Nuclear Security Interfaces.....	13-24
13.5.1 General Nuclear Safety and Security .....	13-24
13.5.2 Security .....	13-24
13.5.3 Physical Security .....	13-25
13.5.4 Cyber Security .....	13-25
13.5.5 Safeguards .....	13-26
13.6 References.....	13-27

**LIST OF TABLES**

None.

## **LIST OF FIGURES**

None.

## **13.0 CONDUCT OF OPERATIONS**

### **13.1 Organizational Structure of Operating Organization**

The prime responsibility for safety is assigned to OPG's operating organization. This responsibility includes covering all activities related to operation directly and indirectly and the supervision of activities of all other related groups, such as design, supply, manufacture and construction, employers, and contractors, as well as the operating organization itself. This responsibility is discharged in accordance with the management system.

This section contains information addressing the following:

1. Design principles used to develop the organizational structure (e.g., layers of hierarchy, length of decision-making chains, scope of managerial control, policy for use of contracted resources)
2. Description of relationships between organizations having significant interaction with information on how any potential effect on nuclear safety management (each relationship) is recognized and addressed
3. Organizational approach taken to ensure capabilities necessary to provide nuclear safety and ensure the integrity of the safety case, including how sufficient in-house core capability is retained to:
  - a. Manage the licenced facility and activities
  - b. Prevent over-reliance on contractors and degradation of in-house capabilities
  - c. Maintain subject matter expertise for all topics necessary for nuclear safety, including "informed (intelligent) customer" roles when expertise is contracted out
  - d. Be an "informed (intelligent) customer" for items or services procured
  - e. Ensure the organization maintains sufficient numbers of qualified workers and identifies nuclear safety-related positions and underpinning roles
  - f. Control organizational changes and maintain the organizational charts as evergreen documents
  - g. Set strategies to ensure the right resources are available at the right time with the right skills and experience to meet core capabilities at all stages of the facility lifecycle and provide for review of implementation and ongoing reviews
  - h. Describe how organizational aspects that lead to vulnerabilities are identified and mitigated (e.g., reliance on scarce or singular areas of expertise)
4. Organization control and how activities will not be subject to undue influence by other organizations
5. Description of how the resource strategy is proactively managed when project work is being implemented to ensure that the resource profiles and organizational arrangements remain fit for the purpose
6. Description of how contracted work is conducted to required levels of safety and quality from an organizational perspective.
  - a. Considerations to address:
    - i. Effective supply chain strategy for delivery of safety case requirements

- ii. “Informed (Intelligent) customer” capability for all work that may affect nuclear safety carried out by contractors or suppliers or their supply chain
- iii. Issuing specifications to contractors or suppliers that adequately describe the items or services, meet the safety case requirements, and identify the required level of Quality Assurance (QA)
- iv. Evaluation and confirmation before placing a contract with nuclear safety significance that the contractors and suppliers have the organizational, technical and project management capability, capacity, and culture to deliver the items or services

OPG manages organizational changes through a reviewed and approved change process.

### **13.1.1 Organizational Structure**

The BWRX-300 is a simple and safe design above traditional Boiling Water Reactors (BWR), requiring a smaller organization while still meeting the requirements of International Atomic Energy Agency (IAEA) Safety Guide NS-G-2.4, “The Operating Organization for Nuclear Power Plants” (Reference 13.6-1). The organizational structure framework is expected to be defined and included with the Licence to Operate application submission with all details, including roles and responsibilities, finalized and in place prior to receipt of the Licence to Operate.

Upper tier management staffing levels are expected to be similar to those at existing OPG facilities. For a single BWRX-300, it is expected that some roles are combined from traditional operating models and could change if additional units are built on the same site.

Staffing levels required to operate the BWRX-300 are expected to be defined based on the safety analysis (with consideration and integration of Human Factors Engineering (HFE)), Maintenance Program, and outage programs as they become better defined. Staff performing operations and maintenance are expected to be qualified as determined using training analysis, using assumptions and findings of HFE analyses. The minimum staffing level complement is expected to be determined in accordance with an analysis performed in accordance with the requirements of CNSC REGDOC-2.2.5, “Minimum Staffing Complement” (Reference 13.6-2).

The Plant Manager is expected to be accountable to OPG management, the Canadian Nuclear Safety Commission (CNSC), and the public to ensure the facility is operated and maintained with due diligence and in a manner consistent with the Power Reactor Operating Licence, and within the social licence objectives set by OPG.

The Operations and Maintenance Manager(s) ensure all aspects of the managed systems for operations and maintenance are implemented. The number of Operations and Maintenance Managers is expected to be defined as the staffing levels and programs are defined. For a single BWRX-300, we propose a single manager who was previously certified or licenced at a nuclear facility. The Maintenance Manager is not required to be previously certified or licenced at a nuclear facility.

The Shift Manager is accountable to ensure the facility is operated within its Operating Licence.

The site organization is augmented with support from the fleet organization, which includes the engineering component. The fleet organization program update framework is expected to be further defined as the design is progressed and included in the Licence to Operate application submission with all programs put in place prior to receipt of the Licence to Operate. Areas in the submission will include:

- Components and Equipment Surveillance

- Major Components
- Equipment Reliability
- Reactor Safety Program
- Aging Management (AM)
- Risk and Reliability
- Chemistry
- Welding
- Environmental Qualification
- Pressure Boundary

#### **13.1.2 Qualifications of Plant Personnel**

The Plant Manager and Operations Manager positions assigned to the BWRX-300 are expected to be filled by staff who have been previously certified or licenced at a nuclear power plant. This could include Small Modular Reactor, BWR, CANada Deuterium Uranium (CANDU), or Pressurized Water Reactor experience.

Qualifications are expected to be developed for each role in the organization according to the Systematic Approach to Training. Role documents defining specific job responsibilities are expected to be developed as appropriate based on the importance of the specific position.

## **13.2 Training**

### **13.2.1 General**

The BWRX-300 training program is developed using a Systematic Approach to Training process that complies with the prescribed regulatory training requirements of CNSC REGDOC-2.2.2, "Personnel Training" (Reference 13.6-3). The OPG personnel training programs ensure worker competence and qualification to perform the duties of their positions.

The OPG training system is developed and implemented to adhere to two fundamental principles:

1. Performance based training is focused on the essential knowledge, skills, and safety attributes required to meet the job requirements (derived from HFE task analysis) and nuclear safety specific needs throughout the lifecycle of the facility.
2. Systematically developed training is defined, produced, and maintained through an iterative and interactive series of steps, leading from the identification and satisfaction of a training requirement.

Training requirements are applied in a manner commensurate with risk. All training requirements apply, but associated training-related processes and procedures may vary based on the safety significance and complexity of the work being performed. The training systems/programs and requirements include:

1. Identification of the performance requirements of a specific job or duty area by conduct of a job task analysis
2. General worker training, initial job training, and continuing training based on a task analysis of the knowledge and skills required to perform each task and any attributes related to safety
3. Training designed, developed, and implemented to meet qualification requirements
4. Trainers meet and maintain documented qualification requirements
5. Formal evaluations used to confirm and document workers are qualified to perform their duties
6. Training change management process that systematically analyzes procedural, equipment, and job description changes (including operational experience feedback) that may require changes to tasks and lead to training modifications
7. Continuing training deemed necessary during the job and task analysis and training needs analyses processes
8. Periodic training program evaluations, with results incorporated into the training improvement process
9. Creation and maintenance of worker training and qualification records
10. Assurance that workers receive the level of training related to nuclear safety that corresponds to their employment and position duties; including, but not limited to radiation safety, conventional safety, fire safety, and on-site emergency arrangements

In addition, training programs are established for initial personnel certification and maintenance of regulatory certifications. Initial and continuing certification training programs are implemented in accordance with the principles of a Systematic Approach to Training.

Positions requiring regulatory certification are expected to be defined based on the technology needs and safety significance. Certification programs are expected to be developed as part of

the design process and only one certification program is required to meet the regulatory requirements.

### **13.2.2 Training Managed System Plan**

All training of personnel is expected to be designed, developed, and delivered using a Systematic Approach to Training.

A full-scope simulator, a replica of the Main Control Room panels, is expected to be utilized to train and qualify control room Operations staff. This approach allows the operators to interface with the simulated plant system in the Main Control Room environment.

#### **13.2.2.1 Minimum Staffing**

A minimum staff complement program is established to ensure sufficient numbers of qualified workers are present to meet regulatory and facility licence requirements during all credible events in the BWRX-300 safety analysis. The minimum complement staffing numbers are expected to be defined following completion of detailed design and safety analysis, and to be part of the Licence to Operate application submission.

The basis for the minimum staff complement is determined by a systematic staffing analysis as described in Chapter 18, Subsection 18.2.5. The analysis to determine the minimum staff complement considers:

1. Actions required in the facility and their timing for the full range of the most resource-intensive conditions
2. Resource-intensive initiating events and credible failures considered in the safety analysis report and the Probabilistic Safety Assessment (PSA) (with HFE considerations)
3. Operating strategies that define how the nuclear facility personnel respond to Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and emergencies
4. Required interactions among facility personnel for the purpose of diagnosing, planning, communicating, coordinating, and controlling AOOs, DBAs, and emergencies
5. Staffing demands required for the possible concurrent use of procedures related to AOOs, DBAs, and emergencies
6. Staffing demands required to monitor indicators, displays, and alarms and to promptly and effectively operate the facility's equipment controls using procedures related to AOOs, DBAs, and emergencies
7. Staffing demands required to perform tasks in field locations using procedures related to the events considered within the scope of the analysis
8. Staffing demands required for the successful completion of any important human actions using procedures related to the events considered within the scope of the analysis
9. Restrictions on the location of workers within the nuclear facility

The minimum staff complement requirements are validated to provide assurance that there are sufficient numbers of qualified workers available to operate the facility safely and respond to the most resource-intensive conditions at all times. Validation of the minimum staff complement is in accordance with the verification and validation processes described in CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 13.6-4).



NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

The minimum staffing requirements are expected to be formalized in a procedure that describes:

1. The specific number of staff to be present on-site, in the facility, and in the Main Control Room, and the composition of the minimum staff complement with reference to specific positions or qualifications
2. Modifications to minimum staff complement for different operational states and the specific number and composition of the minimum staff complement with reference to specific positions or qualifications for each operational state
3. Any specific restrictions on the location of individuals in the facility
4. Measures in place to monitor compliance with the minimum staff complement and to prevent non-compliance with the minimum staff complement
5. Specific actions to be taken to reduce the risk to the facility in the event of non-compliance with the minimum staff complement

### **13.3 Implementation of the Operational Safety Program**

#### **13.3.1 General Implementation**

The OPG Managed System is implemented under Nuclear Management System N-CHAR-AS-0002 (Reference 13.6-5). The system is implemented by a series of program documents which in turn define the required implementing procedures and standards. The Managed System is designed to be fully compliant with CSA Group (CSA) N286, "Managed System Requirements for Nuclear Facilities" (Reference 13.6-6). As such, all implementing procedures and standards ensure that all aspects of CSA N286 are fulfilled, as well as being fully compliant with all CNSC REGDOCs. The Managed System framework associated with BWRX-300 plant operation is expected to be outlined and included with the Licence to Operate application, with all programs finalized and in place prior to receipt of the Licence to Operate.

Changes to the Managed System are made in accordance with Nuclear Management System Administration, N-PROG-AS-0001 (Reference 13.6-7). Changes to the current charter (N-CHAR-AS-0002) (Reference 13.6-5) are expected to be required to accommodate BWRX-300 technology.

#### **13.3.2 Conduct of Operations**

This section addresses important operational issues relevant to safety throughout the lifetime of the plant and how the operating organization addresses identified issues adequately.

The OPG Nuclear Management System sets the standards for health, safety, environment, security, economics, and quality during facility design, construction, commissioning, and operation based on the authority of and a safety culture driven by the OPG Nuclear Safety Policy. The OPG Nuclear Management System promotes the safety culture by committing workers to adhere to the OPG Nuclear Management System, implementing practices that contribute to the excellence in worker performance, supporting workers in carrying out their tasks safely and successfully, and monitoring to improve the culture. The organizational structure implements the programs that make up the OPG Nuclear Management System with the Chief Nuclear Officer accountable for implementation and effectiveness of the OPG Nuclear Management System. The outline of the programs and standards utilized for operating the plant is expected to be included with the Licence to Operate application submission, with all program details and standards finalized prior to receipt of the Licence to Operate.

The OPG Nuclear Management System is based on a set of principles implemented in a graded approach consistent with CNSC REGDOC-2.1.1, "Management System" (Reference 13.6-8) and CSA N286 (Reference 13.6-6) guidelines.

##### **13.3.2.1 Safety Culture**

Consistent with CNSC REGDOC-2.1.2 "Management System: Safety Culture" (Reference 13.6-9), the safety culture is established, promoted, communicated, and fostered by Senior Management through the OPG Nuclear Safety Policy and OPG Nuclear Management System. The safety culture is applicable to all activities that affect the health and safety of workers, the public, and the environment in every phase of the facility life cycle.

The OPG Nuclear Management System is maintained in accordance with the requirements of CSA N286 (Reference 13.6-6).

The OPG Nuclear Management System meets the CNSC's Safety and Control Area principles and regulatory requirements necessary to protect health, safety, and the environment.

The safety culture is implemented, monitored, and periodically assessed consistent with CNSC REGDOC-2.1.2 (Reference 13.6-9) and CSA N286 (Reference 13.6-6) guidelines through

subordinate policies, programs, processes, and procedures that implement the varied administrative, maintenance, and operational aspects of facility operation. An established program, that summarizes OPG's internal and external processes used for oversight and assessment, tracks assessment action items and monitors various metrics that may reveal safety culture aspects (e.g., Operating Experience (OPEX), performance trends, condition reports, regulatory inspections). OPG's Human Performance and Performance Improvement programs also implement OPG expectations for understanding and promoting a strong safety culture.

A more detailed discussion of the OPG Nuclear Management System is provided in Chapter 17, Sections 17.1 and 17.2.

#### **13.3.2.2 Services and Equipment Acquisition/Receipt**

The OPG supply chain process is established and controlled. OPG supply chain services are responsible for establishing and maintaining an OPG nuclear approved supplier list. Periodic audits are performed to confirm the initial and ongoing acceptability of the supplier's management system. The OPG process/program describes methods used to originate, request, evaluate, qualify, and maintain qualification of suppliers of items and services required for QA programs or other OPG nuclear quality requirements.

Suppliers are assessed on their ability to meet purchasing requirements and have the organizational, technical and project management capability, capacity, and culture to deliver the item or service.

The supplier-customer relationship is monitored to ensure purchasing requirements are met. Monitoring includes the alignment of demand and supply signals between OPG and the supplier, a supplier/customer performance assessment, involvement of the supplier in customer demand planning, reporting requirements for delays or defects and supplier involvement in obsolescence and remediation. Monitoring results are used as input in determining the frequency and extent of inspection, verification, and audit activities. The audit program is established as part of the QA Program and is discussed further in Chapter 17, Sections 17.2 – 17.4.

OPG specifies the requirements for purchased expertise and equipment, provides work oversight, and technically reviews the output before, during, and after implementation. Contractors within the supply chain are also audited on a regular basis as part of the contractual agreements.

Components are checked when initially received to ensure the components are as ordered, undamaged, and are not fraudulent, counterfeit, or suspect. The components are subjected to more detailed inspection for acceptability prior to use. After receipt, the components are stored to protect against construction activities, physical and environmental damage, and deterioration.

#### **13.3.2.3 Fitness for Service**

The fitness for service safety and control area covers activities that affect the physical condition of Structures, Systems, and Components (SSC) to ensure adequacy and ability to perform their intended functions when required. Fitness for service is addressed in established programs that include Reliability, Maintenance, AM, Chemistry Control, Periodic Inspections, and In-Service Inspections.

Programmatic requirements addressing fitness for service span the full life cycle of the facility, beginning with inclusion in facility design decision-making and consideration during each phase (e.g., design, construction, commissioning, operation) of the facility's life. Requirements evolve as the facility ages and specific process requirements may vary based on the life cycle phase (e.g., construction versus operation).

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

Reliability is incorporated during facility design, consistent with the requirements of CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 13.6-10) and through the Reliability Program that complies with CNSC REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants" (Reference 13.6-11). The Reliability Program is implemented to ensure that systems function reliably in accordance with design and performance criteria. Although the Reliability Program focuses primarily on the facility operational phase, it applies to all phases of the facility life cycle. The Reliability Program includes:

- Identification and categorization of systems using a systematic process
- Identification of specific failure modes and specification of reliability targets
- Specification of minimum capability and performance level consistent with safety targets and regulatory requirements
- Provisions for information incorporation into maintenance programs
- Provisions for inspection, tests, modeling, and monitoring to assess reliability based on safety class
- Documentation of program activities, attributes, elements, results, and administration

The facility Maintenance Program establishes a maintenance strategy, based on the plant design and safety analysis, to ensure that SSC function as designed. The facility Maintenance Program is implemented by a maintenance organization, established consistent with CNSC REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants" (Reference 13.6-12). A systematic approach is used to identify the SSC maintenance activities to be performed and the maintenance intervals.

The Maintenance Program describes the processes for planning, monitoring, scheduling, and executing maintenance work activities, including those maintenance activities performed during the construction and commissioning phases. Surveillances conducted as part of the Maintenance Program, including acceptance criteria, are addressed in Chapter 16, Sections 16.2 and 16.4, Operational Limits and Conditions.

An AM Program conforming to the requirements of CNSC REGDOC-2.5.2 (Reference 13.6-10) and CNSC REGDOC-2.6.3, "Aging Management" (Reference 13.6-13) is established to ensure the reliability and availability of the required SSC safety functions throughout the facility service life.

The effects of aging and wear are taken into consideration during the design of Safety Class SSC. The considerations include:

1. Design margin assessment that considers the known aging and wear mechanisms potential degradation in operational states, to include the effects of testing and maintenance
2. Provisions for monitoring, testing, sampling, and inspecting SSC to assess aging mechanisms and identify degradation that may occur during operation a result of aging and wear
3. Online monitoring to provide forewarning of degradation leading to failure and where failure could be safety significant

Details regarding AM design provisions are provided in Chapter 3, Subsection 3.1.12.

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

Chemistry Control policies and goals are established to:

- Preserve the integrity of SSC
- Minimize the effects of chemical impurities and corrosion on SSC
- Implement As Low as Reasonably Achievable to manage radioactive material buildup
- Limit release of chemicals and radioactive material to the environment

Chemistry Control governs the development and maintenance of chemistry procedures, specifications, and methods of control. Knowledgeable and trained staff are assigned to monitor for abnormal trends so that action can be taken to ensure operations within specified limits. Performance indicators are maintained to satisfy reporting requirements.

Included in Chemistry Control are requirements for:

- Data management (to include trending, evaluation, and reporting of analysis results and investigations)
- Chemistry surveillance program
- Chemistry specifications for systems
- Procedures for chemistry parameter selection, monitoring, analysis, and trending
- Procedures for operations involving chemistry processes and evaluation of results
- Operation and reference limits for chemistry parameters and associated action levels
- Chemical Control Program
  - Training (chemical hazards, labeling and storage)
  - Procedures for the storage and handling of chemicals
  - Approval, procurement, and receipt of chemicals
  - Listing of chemicals approved for site use and those that are precluded from site use or other classification
  - Administrative controls for controlling products in the workplace

The Chemistry and Chemical Control programs, as applicable to construction and commissioning, are described in Chapter 14, Subsection 14.2.4.

Periodic and in-service inspection and testing programs are established in conformance with CNSC REGDOC-2.5.2 (Reference 13.6-10) to confirm that service-induced degradation has not increased the likelihood of a failure of a barrier against the release of radioactive material.

Periodic and in-service inspection and testing are established for:

- Nuclear pressure boundary components
- Containment components
- Containment structures
- Safety-related structures
- Balance-of-plant pressure boundary Safety Class components or based on AM requirements

#### **13.3.2.4 Nuclear Material Packaging and Transport**

Processes and procedures are expected to be established that address the safe packaging, registration, and transport of nuclear substances to and from the facility as described in OPG's Radioactive Material Transportation Program. The program ensures shipping packages are designed and maintained to ensure protection and containment of the quantities of nuclear material transported. In addition, package certification, package testing, inspection, and maintenance are addressed within the program. This program is expected to be established prior to fuel delivery to the Darlington site, an activity which will be subject to separate licensing by CNSC, as the Licence to Construct scope does not include transport, import, possession, or storage of nuclear fuel.

#### **13.3.3 Maintenance, Surveillance, Inspection and Testing**

This section provides a description and justification of arrangements that the operating organization has in place to identify, control, plan, execute, audit, and review maintenance, inspection, and testing practices that influence reliability and affect nuclear safety.

SSC credited in the safety analysis are identified and periodically tested (surveilled) at a frequency related to the results of reliability analysis and operational experience to ensure that they will function as required. SSC performance that is inconsistent with assumptions in the safety analysis are identified. Following modification to SSC, the test requirements are re-evaluated. Defense Line 3 (Safety Category 1) SSCs credited in the deterministic safety analysis are addressed in the Operational Limits and Conditions (OLC) of Chapter 16 and the Defense Line 2 and 3 SSCs credited in the deterministic safety analysis are addressed in a program required by Chapter 16. Furthermore, additional SSCs credited in the probabilistic safety analysis will be addressed in the OLC if its failure is a significant contributor to Core Damage Frequency. This defence-in-depth approach provides reasonable assurance the consequences of postulated initiating events are bounded by Chapter 15 results and safety goals are met

SSC are maintained in accordance with a maintenance strategy defining the frequency and type of maintenance to be performed, taking into consideration the supplier recommendations, safety analysis, periodic inspection requirements, OPEX, cost benefit analysis, and service conditions. Maintenance activities are performed in accordance with approved procedures and practices. Preventive measures are employed to eliminate structural, system, and component damage or the contamination of systems with foreign material. In addition, predictive maintenance is performed based on plant monitoring system information. Maintenance includes the repair or replacement of malfunctioning SSC as needed to re-establish conformance with requirements.

A Maintenance Program is to be implemented consistent with the requirements of CNSC REGDOC-2.6.2 (Reference 13.6-12) to address:

1. Measures, policies, methods, and procedures providing direction for maintaining SSC capable of maintaining their functions as described in design documents and the safety analysis
2. Processes for planning, monitoring, scheduling, and executing work activities so SSC continue to meet design intent and remain fit for service in the presence of degrading mechanisms
3. Preventive maintenance activities, maintenance processes and record retention requirements, corrective maintenance, calibration of measuring and monitoring devices, SSC monitoring (activity optimization), outage management, work planning and scheduling, work execution, maintenance procedures, post-maintenance verification and testing and Maintenance Program assessment

4. Predictive maintenance based on plant monitoring system information
5. Surveillance program covering OLC, with surveillance frequencies based on a reliability analysis, a PSA, and previous OPEX, and that shows viability of inspection techniques to meet performance requirements while taking As Low as Reasonably Achievable into account
6. Approach taken to develop SSC surveillance program acceptance criteria
7. Assurance that the surveillance program is adequate to ensure the inclusion of all relevant aspects of the OLC
8. Timeline for the development of each program with milestones for development and implementation of each program and the processes followed
  - a. Results of each activity to be reviewed against acceptance criteria and with periodic reviews to ensure the program continues to meet objectives.

Multiple aspects of the surveillance, inspection, and testing program are addressed within OLC (Chapter 16, Section 16.4), to include:

1. Safety Class plant items that require monitoring to ensure they remain fit for their purpose and operation is within the operational limits for reliable and safe operation
2. How surveillance, maintenance and repair ensure OLC parameters remain within acceptable limits and systems/components are operable
3. Surveillance frequency basis on a reliability analysis, including where available, a PSA and a study of experience gained from previous surveillance results (in the absence of both, the surveillance is based on supplier recommendations)
4. System for ensuring testing is performed and confirmed within the timelines allowed

### **13.3.4 Core Management and Fuel Handling**

#### **13.3.4.1 Core Management**

The programs and procedures that govern the operational activities associated with BWRX-300 core management regarding fuel reliability are based on guidelines established by GEH, utilizing decades of experience with fuel from Global Nuclear Fuel. Fuel related design aspects, including operational, transient, and accident limits, are discussed in Chapter 4, Sections 4.2 and 4.4. The core/fuel management guidelines are implemented through operational methods implemented to mitigate and reduce duty related fuel performance risks.

In general, the BWRX-300 operational methods employ an approach of limiting the duration of low power periods and limiting the rate at which power is raised following prolonged operation at low power. When raising power, a combined approach of an unrestricted power increase to an established threshold or prior conditioned power envelope, followed by raising power to a final value at a defined, controlled, slow ramp rate, is used.

The operational practices are based on BWRX-300 operational experience:

1. An established exposure dependent Linear Heat Generation Rate threshold, below which no power maneuvering restrictions are applied with power increases above the threshold limited to a defined controlled rate
2. Power envelopes (conditioned power) established by the maintenance of specific power conditions sustained for a defined period

3. Defined power ramp rates for power increases above the more limiting of a Linear Heat Generation Rate threshold or the conditioning envelope value, performed at a defined, controlled ramp rate
4. Sequence exchange intervals established based on cycle exposure, that consider multiple factors, including fuel reliability
5. Threshold power levels established for fuel bundles or nodes with unusually long periods of low power operation (long control intervals), implemented on a case-by-case basis using industry OPEX best practices
6. Employing power envelopes best practices considering the BWR characteristics of top peaked axial power shapes at the end of fuel cycle and bottom peaked axial power shapes at the beginning of fuel cycle operation
7. Control rod exercising requirements
8. Barrier fuel risk mitigation
9. Established threshold values for fuel with high residence time in central portions of the core

Core Monitoring is a function of the plant computer system that provides three-dimensional core power monitoring to satisfy the requirements of CNSC REGDOC-2.5.2, Section 8.1 (Reference 13.6-10) to ensure the plant operates within the power distribution design basis. Core Monitoring provides confidence that the plant is operating in conformance to specified acceptable fuel design limits. Core Monitoring obtains instrumentation information from the C20 Distributed Control and Instrumentation System (refer to Chapter 7, Subsection 7.3.3.2), calculates thermal power limits, and provides estimates of power distributions. These estimates are calculated by the core simulator.

The Core Monitoring function acquires real-time reactor data from site plant data acquisition systems as necessary to define the reactor state for use by the core simulator. Core Monitoring can calculate the accumulated thermal and electrical energy produced by the plant from the beginning of an operating cycle. The Core Monitoring function is described further in Chapter 4, Subsection 4.6.8.

### **13.3.5 Aging Management and Long-Term Operation**

This section provides a description of an integrated AM Program that will meet the requirements of CNSC REGDOC-2.6.3 (Reference 13.6-13) and CNSC REGDOC-2.5.2, Section 7.17 (Reference 13.6-10). AM processes and plans ensure the reliability and availability of required safety functions of SSC throughout the service life of the facility (Lifecycle Management Plans). Periodic inspection or in-service inspection programs, as they relate to BWRX-300 aspects, are expected to be incorporated directly into AM programs.

AM is addressed during design within the design process. The design provisions for AM are discussed in more detail in Chapter 3, Section 3.1. Consideration is given to the feedback of OPEX. A systematic approach is taken during design to understand the aging of SSC to evaluate design features for aging prevention, monitoring, and mitigation. Mechanical, thermal, chemical, electrical, physical, biological, and radiation aspects are taken into consideration. SSC determined to have shorter service lives than the nominal design life are identified with AM strategies provided in the design documentation. The components that are identified with service lives less than the nominal design life also have replacement plans defined in the plant Maintenance Program, with associated monitoring requirements and provisions to permit their removal and replacement.



NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

Aging effects under design basis conditions, including transient and postulated initiating event conditions, are also considered in equipment qualification programs.

The design information derived with these design aging considerations establish the baseline for the test data required to be collected and documented for AM Program monitoring and evaluation requirements.

Design documents also identify any special manufacturing or construction processes that are to be applied to prevent, mitigate, or eliminate known aging mechanisms. These provisions are necessary for specification in procurement documents.

The AM Program and processes are used to detect, assess, and manage deterioration of SSC as a result of aging effects such as irradiation, corrosion, erosion, fatigue, and other material degradation.

Descriptions of the following AM Program elements include:

- Organizational arrangements
- Data collection and record keeping
- Screening and selection process for AM
- Evaluations for AM
- Condition assessments
- SSC-specific AM plans
- Management of obsolescence
- Interfaces with other supporting programs
- Implementation of SSC-specific AM plans
- Review and improvement processes for the AM Program

An integrated AM Program ensures that availability and reliability of required safety functions throughout the facility's service life is established. The program requires AM activities to be implemented proactively throughout the life cycle of a nuclear facility in compliance with CNSC REGDOC-2.6.3 (Reference 13.6-13).

### **13.3.6 Control of Modifications**

This section addresses the method of identification for designing, planning, executing, controlling, testing, auditing, reviewing, and documenting modifications to the plant throughout its lifetime, consistent with the guidance provided in IAEA SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 13.6-14). The modification control process covers all safety significant changes (permanent and temporary) made to SSC, OLC, plant procedures, and process software. The design and safety analysis is incorporated into the purchasing, construction, commissioning, operating, and maintenance documentation such that the as-built configuration of the facility is aligned with the design and safety analysis. Design authority configuration requirements, including the responsibilities and authority of organizations whose functions affect the configuration of the facility, including activities such as design, maintenance, construction, licensing, and procurement, are controlled through its Configuration Management System. A series of programs, including engineering change control, design management, and software, ensures plant configuration is controlled in a manner that is analyzed to be safe. Control of modifications and configuration management during construction and commissioning phases is discussed further in Chapter 14, Section 14.1.

The information provided includes descriptions of:

1. Modification control process for maintaining the design basis, taking into account new information, OPEX, safety analyses, resolution of safety issues, or correction of deficiencies
2. Description of how design changes are assessed, addressed, and accurately reflected in the safety analyses or record prior to implementation

The plant modification control process covers:

1. Changes made to plant systems and components, OLC, plant procedures and process software, taking into account the safety significance of the proposed modifications to allow them to be graded and referred to the CNSC when necessary
2. Changes to task performance requirements (task step alterations, expected outcomes, procedure level), personnel job role responsibilities or the operating organization
3. Records retention, and where necessary, revision documentation, procedures, instructions, and drawings to reflect the changes

#### **13.3.6.1 Configuration Management**

Configuration management is incorporated into purchasing, operating, and maintenance so that the as-built configuration of the facility aligns with the design and safety analysis in accordance with CSA N286 (Reference 13.6-6) and CSA N286.10, "Configuration Management for High Energy Facilities" (Reference 13.6-15). Configuration management is applied in a graded approach.

Configuration management during the construction and commissioning phases is described further in Chapter 14, Section 14.1.

Configuration management is not a stand-alone program. Configuration management plans are developed and integrated within the OPG Nuclear Management System (e.g., assessment, problem identification and resolution, training). From conception to the end of operations, configuration management ensures data generated during design, construction, and commissioning reflects the design basis and specified requirements are kept current in the design, as-built, and field change documentation.

The design basis and requirements for the BWRX-300, including safety analysis, are established and documented in accordance with CSA N286 (Reference 13.6-6) and are traceable to the respective SSC. Impacts of design changes are assessed, addressed, and when applicable, reflected in the safety analysis. Subsequent changes to the physical and operational configuration are maintained consistent with design requirements and configuration information throughout the operational life cycle. Where SSC requirements exceed functional design requirements (safety margin versus design margin), the process ensures safety margin is maintained for subsequent modifications. Physical assessments of SSC configuration are conducted as part of facility management.

Temporary and permanent changes are managed in accordance with the requirements of CSA N286 (Reference 13.6-6).

Configuration information, the types and sources of configuration information, and associated documentation are controlled and maintained in accordance with CSA N286 (Reference 13.6-6), with the status of changes identifiable. As-built information is turned over prior to commencement of operations (turnover) and subsequent as-built information is incorporated in a timely manner commensurate with the associated risk. The facility design basis is maintained following turnover

and commissioning by OPG to reflect new information, OPEX, safety analyses, and the resolution of safety issues or deficiency corrections.

Configuration deviations, when identified, are managed through the problem identification and resolution processes consistent with the requirements of CSA N286 (Reference 13.6-6). Deviations are immediately controlled (if required), documented, evaluated for significance, and the underlying cause assessed if deemed systemic and accepted. Problem resolutions are reviewed for effectiveness.

Configuration management objectives and concepts are addressed in the respective training programs, with the necessary links between configuration management and the training programs established and maintained.

Configuration information records and documents are maintained consistent with the requirements of CSA N286 (Reference 13.6-6).

#### **13.3.6.2 Engineering Change Control**

OPG engineering change control is an integrated management process that ensures the physical and operational configuration and documentation continue to conform to the design and licensing basis requirements.

Facility configuration is maintained from initial fuel load to the end of operating life through established programmatic configuration and change control processes that adhere to CSA N286 (Reference 13.6-6).

The change control process makes certain that safety limits, design basis, licensing basis, and normal operating margins are controlled under engineering change control to always ensure the facility is operated well within conditions analyzed to be safe. Additionally, this process ensures all changes, from minor parts substitution to safety-related modifications, are controlled to ensure the designed facility is operated with margin.

#### **13.3.6.3 Design Management**

The Design Management Program will specify requirements for the following two areas:

1. Management of prescribed activities appropriate for execution and control of required design, design support, and documentation for nuclear facilities and organizations in accordance with CSA N286 (Reference 13.6-6)
2. Processes for creating or modifying documentation required for controlling design bases and design outputs

A minimum set of documentation identifies and describes design bases, design output, and the design process.

#### **13.3.6.4 Software**

The Software Program complies with CSA N286 (Reference 13.6-6) and CSA N286.7 "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 13.6-16), and applies to software classified as Real-Time Process Computing and Scientific, Engineering, and Safety Analysis Software and Software Engineering Tools. The program identifies the processes and overall requirements for classification of software and identifies governing standards for each software classification, defining requirements for software development, maintenance, procurement, qualification, use, and retirement.

### **13.3.7 Program for the Feedback of Operating Experience**

This section describes the program implemented for the feedback of OPEX. The OPEX Program ensures operational events and incidents occurring at the facility and other relevant facilities are captured or identified, recorded, notified, investigated internally, and used to incorporate lessons learned for the operation of the facility.

Relevant OPEX is considered for the BWRX-300 during design, construction, commissioning, operation, maintenance, and decommissioning. The design authority (GEH) establishes provisions for the incorporation of OPEX through Integrated Management Systems. The OPEX comes from a variety of sources, including direct input, GEH/Global Nuclear Fuel experience from operating the BWR and Advanced Boiling Water Reactor fleet, Institute of Nuclear Power Operations, Electric Power Research Institute, Boiling Water Reactor Owners' Group, U.S. Department of Energy, U.S. Nuclear Regulatory Commission, CANDU Owners Group, and CNSC. OPEX associated with the construction and commissioning phases is discussed in Chapter 14, Section 14.1.

Industry OPEX information is routinely made available to or distributed by GEH design and modifications personnel. The more important industry-wide issues are routinely addressed in CNSC Nuclear Incident Reports and U.S. Nuclear Regulatory Commission Generic Letters and Bulletins.

OPG has an established OPEX process for evaluating, integrating, accessing, and sharing OPEX information. The OPEX process addresses implementation of OPEX feedback during design activities and its continuance through the construction, commissioning, and operational phases of the facility life cycle, to include how events are identified, recorded, investigated, and reported; as well as how findings from the events are used to enhance safety performance.

### **13.3.8 Documents and Records**

This section addresses the programmatic provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising, and deleting documents, records, and reports relevant to the operation of the facility over its lifetime in accordance with OPG-PROG-0001 "Information Management" (Reference 13.6-17).

Document and records program management is the responsibility of the operating organization.

OPG records management encompasses the control of documents and records with requirements addressed in the Controlled Document Management Program. The OPG process for the control of documents includes the development, validation, and approval of safety-related documents. Documents are available for use at the location where the work is to be performed. Changes to documents are documented and tracked. The OPG process for the control of records ensures that records are readable, complete, identifiable, traceable, retrievable, preserved, and retained as necessary.

Documents are controlled consistent with their intended use and consistent with CNSC REGDOC-2.1.1 (Reference 13.6-8) and CSA N286 (Reference 13.6-6).

The program document ensures that controlled documents include:

- Unique identification
- Defined format and presentation
- Identification of status
- Review for adequacy and approval

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

- Availability for use at the location where the work is performed or where the document is required for reference
- Prompt removal of obsolete documents for use

Records are:

- Readable
- Complete
- Identifiable
- Traceable to the related items and work
- Retrievable
- Preserved
- Retained as specified

Document management for the BWRX-300 is controlled under the QA Program during design. The QA Program includes document management aspects. The QA Program requirements during design of the BWRX-300 are established in NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 13.6-18). NEDO-11209-A includes requirements during design, addressing:

- Procurement document control
- Instructions, procedures, and drawings
- Document control
- Control of QA records

Document and records management is discussed with respect to the QA Program in Chapter 17, Section 17.2. Documents and records management during construction and commissioning is discussed in Chapter 14, Section 14.1.

### **13.3.9 Outages**

This section addresses the programmatic aspects of the conduct of periodic reactor shutdowns (outages).

The current reference cycle for DNNP BWRX-300 is based on a nominal 12-month fuel cycle. Different fuel cycle durations can be supported depending on the overall fuel reload strategy to be deployed on a cycle/multi-cycle specific basis.

Outage analysis does not address forced outages, but surveillance and maintenance activities that require the plant to be shut down are minimized to the extent possible, largely by enhanced system reliability achieved through design simplicity.

Outage planning, scheduling, and maintenance activities are managed consistent with the guidance provided in CNSC REGDOC-2.6.2 (Reference 13.6-12), and reporting requirements are consistent with the requirements of CNSC REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants" (Reference 13.6-19).

The Work Management Program provides for the implementation of processes and procedures for the planning, scheduling, and execution of maintenance activities. Work planning is conducted at both the overall plant and individual job levels. The Outage Management Program establishes the criteria followed to confirm that planned outage and emergent work is completed satisfactorily.

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

In addition to procedures for routine outage maintenance activities, OPG maintains a forced outage plan for emergent conditions.

Outage plans are reviewed for nuclear safety, with work groups reviewing the plans within their area of responsibility and with specific consideration given to:

- Impact on operating units and systems
- Application of controls during infrequently performed tests and evolution to ensure the plant is maintained within the design basis
- Contingency plans for alternate measures to maintain safe shutdown
- Routine review to capture changes from the original plan impact assessment
- Outage OPEX

The cumulative effect of plant equipment taken out of service is taken into consideration to ensure there are no adverse effects on the performance of safety functions when planning and scheduling outage work. In addition, plans to remove equipment from service during an outage include measures to deal with the possible consequences of an event occurring while the equipment is out of service. Clear statements are made to identify when equipment is being taken out of service, to include the duration and impact of removing the equipment from service.

The outline of the Outage Management Program utilized for operating the plant is expected to be included with the Licence to Operate application, with all details finalized prior to receipt of the Licence to Operate.

### **13.4 Plant Procedures and Guidelines**

This section programmatically addresses the relevant documents used by plant staff to ensure that procedures and guidelines for normal operation, AOOs, and accident conditions are followed in the intended manner. Procedure development is a technical element of the BWRX-300 HFE program. The procedure development process is described in Chapter 18, Subsection 18.3.7.

#### **13.4.1 Administrative Procedures**

This section describes the administrative procedures that outline the essential elements of the administrative programs used by the operating organization to ensure the safe management of the plant. The processes to develop, approve, revise, and implement the procedures are described along with a list of the relevant procedures.

Administrative procedures contain adequate programmatic controls to provide an effective interface between organizational elements. This includes contractors or organizations providing support to the facility operating organization.

Procedure Writer's Guides promote standardization and the application of HFE usability principles to administrative procedures. Additional details are provided in Chapter 18, Subsection 18.3.7.

Procedural compliance with all administrative procedures ensures all regulatory requirements and standards are met. Procedural steps that implement these specific requirements are flagged with "bases" statements (e.g., [B-1] meaning refer to B-1 for the overriding regulatory and legal requirement). The "Content Authority" is accountable to ensure the administrative procedure meets applicable regulatory requirements and standards. The flagging of bases requirements ensures that regulatory and legal requirements are checked during the continuous improvement (revision) process.

Procedure maintenance and control of procedure updates are performed in accordance with OPG's QA Program processes.

The plant administrative procedures provide for the following:

- Establishment of a formal review and approval process
- Control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment
- Control of maintenance and modifications
- Temporary changes
- Temporary procedure issuance and control
- Special orders of a temporary or self-cancelling nature
- Standing orders to shift personnel, including the authority and responsibility of the control room staff
- Manipulation of controls and assignment of shift personnel to duty stations
- Shift relief and turnover procedures
- Fitness for Duty
- Working hour limitations
- Feedback of design, construction, and applicable important industry OPEX
- Shift Manager administrative duties

- Verification of correct performance of operational activities
- Vendor interface program that provides vendor information for SSC is incorporated into plant documentation
- Fire protection program implementation
- Process for implementing safety/security interface requirements

#### **13.4.2 Operating Procedures**

The facility is operated, monitored, and maintained within the safe operating envelope and in accordance with procedures that are consistent with the design. Operating procedures are established to provide for the safe conduct of BWRX-300 normal operations. Normal operation is operation within specified OLC, within one of the following plant operating modes (further defined in Chapter 16, Appendix 16A):

- Mode 1: Power Operation
- Mode 2: Startup
- Mode 3: Hot Shutdown
- Mode 4: Stable Shutdown
- Mode 5: Cold Shutdown
- Mode 6: Refueling

Procedure Writer's Guides promote standardization and the application of HFE usability principles to the operating procedures. Additional details are provided in Chapter 18, Subsection 18.3.7.

Normal, abnormal, unplanned, and emergency operating procedures are validated to be accurate and usable without any human error traps and verified to be consistent with the safe operating envelope.

Plant operations are performed in accordance with procedures, with use and adherence direction provided for the worker. Temporary procedures may be issued when existing permanent procedures are not applicable to the work being performed. Temporary procedures are periodically reviewed for applicability and cancelled when no longer required.

Operating procedures address:

- Normal operation
- Abnormal operation
- Emergency operation
- Refueling and outage planning
- Alarm response
- Maintenance, inspection, test, and surveillance
- Beyond design basis and severe accidents

The status of SSC is controlled with the following requirements:

- Status changes must be authorized
- Operational position of Safety Class devices is known and controlled



- Status of SSC under maintenance, inspection, or test is known
- Field equipment deficiencies are identified
- Placement and removal of tags on systems and components (e.g., caution tags, work protection tags, terminal point tags, and other similar tags) is controlled
- Plant status information is transferred during shift turnovers

### **13.4.3 Procedures and Guidelines for Operating the Plant During Accidents**

#### **13.4.3.1 Emergency Operating Procedures**

This section describes the programmatic approach followed to develop the Emergency Operating Procedures (EOPs) in accordance with CNSC REGDOC-2.3.2, "Operating Performance – Accident Management" (Reference 13.6-20), and procedure development that supports the operator when responding to anticipated and unanticipated events. EOPs are developed in accordance with CNSC REGDOC-2.5.2, Section 4.2.4 (Reference 13.6-10).

Emergency procedures are available for non-routine and emergency conditions that require immediate action. Emergency conditions addressed include unexpected radiological and non-radiological hazards, excessive emission of radiological and non-radiological liquid or gaseous effluent, fires, and natural disasters. Emergency procedures are kept in prominent, easily accessible locations. Emergency procedures are exercised in practice drills to ensure that requirements are met.

EOPs implement the strategies and measures employed in the integrated accident management plan and ensure that escalation of an accident is avoided, the accident progression is terminated, and fission product releases are kept to a minimum. The EOPs contain a set of information, instructions, and actions designed to prevent the escalation of an accident, mitigate its consequences, and bring the reactor to a safe and stable state. The development of these procedures takes into consideration the information available to the operating staff and conditions where some of the information may be incomplete with significant uncertainties. Also taken into consideration are long time periods to initiate and complete required actions, human and organizational performance, and the possibility of prolonged times to restore power.

All EOPs are developed in accordance with a systematic procedure development plan that considers HFE principles in both the actions required by the procedure and the design of the procedure itself. Procedure development is a technical element of the BWRX-300 HFE Program, as described in Chapter 18, Subsection 18.3.7.

#### **13.4.3.2 Guidelines for Accident Management**

This section describes the programmatic approach followed to develop accident management procedures and guidelines, including EOPs, Emergency Mitigating Equipment (EME) guidelines and Severe Accident Management Guidelines (SAMGs) in accordance with CNSC REGDOC-2.3.2 (Reference 13.6-20).

Accident management includes multiple components such as equipment and instrumentation, procedures and guidelines, and organizational accountabilities, and it interfaces with many programs established for a reactor facility. An adequate accident management plan ensures the ability to respond to any credible accident in order to prevent the escalation of the accident, mitigate the consequences of the accident, and achieve a long-term stable state after the accident.

Integrated accident management planning consists of a cohesive set of plans and arrangements undertaken to ensure:

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

1. Safety systems and the available SSC can be used to control the reactivity, cool the fuel, and contain the radioactive materials such that damage to the reactor vessel and harm to workers, public, and environment is prevented or mitigated
2. Personnel with responsibilities for accident management are adequately prepared to utilize the available resources, procedures, and guidelines to perform effective accident management actions and, when deemed necessary, to call for and interact with the emergency response teams

EOPs, EME guidelines, and SAMGs are developed and implemented to facilitate a licensee's capability to manage the AOOs, DBAs, and Beyond Design Basis Accidents, including Design Extension Conditions and severe accidents. These procedures are developed using a systematic approach in accordance with CNSC REGDOC-2.3.2 (Reference 13.6-20) and CNSC REGDOC-2.5.2, Sections 4.2.4, 7.9.3, 8.5, 8.10, and 9.3 (Reference 13.6-10).

The process of accident management planning will define and describe the following requirements:

- Specific goals of accident management
- Requirements of accident management
- Equipment and instrumentation
- Procedures and guidelines
- Organizational accountabilities

A timeline with milestones for the development, validation, and implementation of all operating procedures, EOPs, EME guidelines, and SAMGs for accident management is expected to be provided in the Licence to Operate application submission.

## **13.5 Nuclear Safety and Nuclear Security Interfaces**

### **13.5.1 General Nuclear Safety and Security**

The plans for physical protection of the facility are described in separate, confidential files. This section addresses how safety measures and nuclear security measures are designed and applied programmatically.

OPG, the operating organization, is responsible for managing the implementation of safety requirements and security requirements, with the primary objective of minimizing risk, through programs and processes established to ensure close cooperation between safety managers and security managers. The safety and security measures are designed and implemented through programs and processes in a complementary manner that do not compromise each other. Mechanisms are established within the programs to resolve any potential conflicts and to manage the safety-security interfaces.

### **13.5.2 Security**

The following security measures, in accordance with the Nuclear Safety Regulations (SOR/2000-209, "Nuclear Security Regulations" (Reference 13.6-28)) and consistent with the relevant guidance provided in CNSC REGDOC-2.12.1 Volume I, "High Security Facilities, Volume I: Nuclear Response Force" (Reference 13.6-21); CNSC REGDOC-2.12.1 Volume II, "High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices" (Reference 13.6-22); and CNSC REGDOC-2.12.3, "Security of Nuclear Substances: Sealed Sources and Category I, II and III Nuclear Material" (Reference 13.6-23), are established for the prevention, detection, and response to unauthorized acts, criminal or intentional, that could directly or indirectly produce harmful site consequences:

- Site physical security
- Personnel security
- Information protection
- Document security
- Cyber security

Prescribed and security-sensitive information is only provided to persons with a valid security clearance and "need to know."

Public access to the site-controlled area is restricted by fencing and signage and with OPG Nuclear Security Officer patrols.

OPG maintains a security clearance program consistent with CNSC REGDOC-2.12.2, "Security: Site Access Security Clearance" (Reference 13.6-24). Staff and contractors requiring unescorted access to the site require a security clearance commensurate with activities performed or access required.

Threat risk assessment is performed as part of the Nuclear Security Program, with results taken into consideration in plan development and facility response. A Memorandum of Understanding exists with the Durham Regional Police Services and is maintained to provide for an off-site response to OPG facilities. The agreement(s) ensure that necessary resources are available to address design basis security events. The Memorandum ensures resources are available to address design basis security events in support of existing armed Nuclear Security. OPG periodically conducts drills and exercises that include integrated response with the off-site response force. Lessons learned from drills and exercises are implemented within the Security

Program. The Memorandum is subject to annual review and is revised to reflect existing security postures.

OPG's Cyber Security Program implements the OPG Cyber Security Policy. Information technology and industrial control systems are managed in a secure, vigilant, and resilient manner that minimizes cyber risks to information assets, renewable generation, and nuclear facilities. The Nuclear Cyber Security Program ensures secure operations of computer systems associated with the industrial control systems for OPG nuclear facilities. Cyber security is applied to plant systems, including those used to ensure safe operations and those which provide physical security of OPG nuclear facilities. The Nuclear Cyber Security Program complies with requirements of CSA N290.7, "Cyber Security for Nuclear Facilities" (Reference 13.6-25).

Nuclear Security Officers are selected, trained, and equipped in accordance with the applicable requirements of CNSC REGDOC-2.12.1 Volume I (Reference 13.6-21) and CNSC REGDOC-2.2.4, "Fitness for Duty, Volume III: Nuclear Security Officer Medical, Physical and Psychological Fitness" (Reference 13.6-26).

OPG has programs in place at existing operating nuclear facilities to facilitate compliance with applicable safeguard requirements and agreements. Measures related to site buildings and structures, operational parameters, and the flow and storage of nuclear material throughout the life cycle of the nuclear facility are described in the Environmental Impact Statement.

Details (prescribed information) of the Security Program are transmitted only by secure means consistent with OPG-STD-0030, "Protecting OPG's Information" (Reference 13.6-27) and SOR/2000-202, "General Nuclear Safety and Control Regulations," Sections 21-23, Prescribed Information (Reference 13.6-29).

Details providing security in design, which informs the Security Program, are provided in the Security Annex.

### **13.5.3 Physical Security**

The Nuclear Security Program is implemented by Nuclear Security, N-PROG-RA-0011, "Nuclear Security" (Reference 13.6-30) using a graded approach. The bulk of the program is classified and addressed separately.

### **13.5.4 Cyber Security**

The Cyber Security Program, OPG-PROG-0042 (Reference 13.6-31), procedures and controls ensure the following:

1. Ensure employees and contractors are in compliance with all applicable requirements of this Cyber Security Program
2. A culture of awareness is fostered to promote secure practices in the use of all digital technologies
3. Methods are established to monitor Information Technology and Operational Technology environments on an ongoing basis in order to detect and respond to threats that impact the confidentiality, integrity, and availability of OPG's assets
4. Strategies are in place to prepare for, respond to, and recover from cyber security incidents that impact OPG's reputation, energy production, and public and employee safety

### **13.5.5 Safeguards**

The Safeguards Program, N-PROG-RA-0015 (Reference 13.6-32) is compliant with Nuclear Safety and Control Act, June 2000; its associated General Regulations, and CNSC REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy" (Reference 13.6-33); and includes the following:

- Communication protocol between the IAEA, CNSC, and Ontario Power Generation, Nuclear
- Obligations to meet applicable regulatory requirements and requirements of associated safeguards procedures
- Reporting to meet applicable regulatory requirements and requirements of safeguards agreements

See the Safeguards Annex: Safeguards and Nuclear Material Accountancy for additional details.

### 13.6 References

- 13.6-1 IAEA Safety Standards Series No. NS-G-2.4, "The Operating Organization for Nuclear Power Plants," International Atomic Energy Agency.
- 13.6-2 CNSC Regulatory Document REGDOC-2.2.5, "Minimum Staff Complement," April 2019.
- 13.6-3 CNSC Regulatory Document REGDOC-2.2.2, "Personnel Training."
- 13.6-4 CNSC Regulatory Document REDOC-2.5.1, "General Design Considerations: Human Factors."
- 13.6-5 N-CHAR-AS-0002, "Nuclear Management System," Ontario Power Generation.
- 13.6-6 CSA N286, "Management System Requirements for Nuclear Facilities," CSA Group.
- 13.6-7 OPG Document, N-PROG-AS-0001, Nuclear Management System Administration.
- 13.6-8 CNSC Regulatory Document REGDOC-2.1.1, "Management System."
- 13.6-9 CNSC Regulatory Document REGDOC-2.1.2, "Management System: Safety Culture."
- 13.6-10 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 13.6-11 CNSC Regulatory Document REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants."
- 13.6-12 CNSC Regulatory Document REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants", August 2017.
- 13.6-13 CNSC Regulatory Document REGDOC-2.6.3, "Aging Management."
- 13.6-14 IAEA Specific Safety Guide No. SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," International Atomic Energy Agency.
- 13.6-15 CSA N286.10, "Configuration Management for High Energy Facilities," CSA Group.
- 13.6-16 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 13.6-17 OPG-PROG-0001, "Information Management," Ontario Power Generation.
- 13.6-18 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 13.6-19 CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants."
- 13.6-20 CNSC Regulatory Document REGDOC-2.3.2, "Operating Performance – Accident Management."
- 13.6-21 CNSC Regulatory Document REGDOC-2.12.1, "High Security Facilities, Volume I: Nuclear Response Force."
- 13.6-22 CNSC Regulatory Document REGDOC-2.12.1, "High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices."
- 13.6-23 CNSC Regulatory Document REGDOC-2.12.3, "Security of Nuclear Substances: Sealed Sources and Category I, II and III Nuclear Material," September 2020.

NEDO-33963 REVISION 0  
NON-PROPRIETARY INFORMATION

- 13.6-24 CNSC Regulatory Document REGDOC-2.12.2, "Security: Site Access Security Clearance."
- 13.6-25 CSA N290.7, "Cyber Security for Nuclear Facilities," CSA Group.
- 13.6-26 CNSC Regulatory Document REGDOC-2.2.4, "Fitness for Duty, Volume III; Nuclear Security Officer Medical, Physical and Psychological Fitness," September 2018.
- 13.6-27 OPG-STD-0030, "Protecting OPG's Information," Ontario Power Generation.
- 13.6-28 Government of Canada SOR/2000-209, "Nuclear Security Regulations."
- 13.6-29 Government of Canada SOR/2000-202, "General Nuclear Safety and Control Regulations."
- 13.6-30 N-PROG-RA-0011, "Nuclear Security," Ontario Power Generation.
- 13.6-31 OPG-PROG-0042, "Cyber Security," Ontario Power Generation.
- 13.6-32 N-PROG-RA-0015, "Safeguards and Nuclear Material Accountancy," Ontario Power Generation.
- 13.6-33 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33964

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 14  
Plant Construction and Commissioning**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ASME	American Society of Mechanical Engineers
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
DNNP	Darlington New Nuclear Project
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
OPEX	Operating Experience
OPG	Ontario Power Generation
QA	Quality Assurance
SSC	Structures, Systems, and Components

## TABLE OF CONTENTS

14.0	PLANT CONSTRUCTION AND COMMISSIONING .....	14-1
14.1	Management System (Construction and Commissioning).....	14-1
14.1.1	Configuration Control.....	14-11
14.2	Construction .....	14-15
14.2.1	Organization .....	14-15
14.2.2	Existing Facility Effects .....	14-23
14.2.3	Construction Readiness Review .....	14-23
14.2.4	Construction Program.....	14-26
14.2.5	Work Turnover.....	14-32
14.3	Commissioning .....	14-35
14.3.1	Organization .....	14-35
14.3.2	Commissioning Management .....	14-35
14.3.3	Commissioning Program.....	14-38
14.3.4	Commissioning Tests .....	14-40
14.3.5	Test Phases.....	14-41
14.3.6	Structures, Systems, and Components and Facility Transfer.....	14-42
14.4	References .....	14-43

**LIST OF TABLES**

None.

**LIST OF FIGURES**

Figure 14.1-1: Quality Program Process Documents.....	14-14
Figure 14.3.2-1: Turnover and Commissioning Program Management Plan Framework.....	14-37

## **14.0 PLANT CONSTRUCTION AND COMMISSIONING**

The purpose of Chapter 14, Plant Construction and Commissioning is to describe how OPG assures that the BWRX-300 will be suitable for service prior to entering the construction, commissioning, and operational stages.

### **14.1 Management System (Construction and Commissioning)**

The primary responsibility for safety and security is assigned to the operating organization or the constructor (depending on the project phase). This responsibility includes the supervision of activities of all other related groups, such as design, supply, manufacture, and construction, employers, and contractors, as well as the operating organization itself. This responsibility is discharged in accordance with the OPG management system.

Construction, commissioning, and related activities are developed and implemented under an OPG management system that meets the requirements of CSA Group (CSA) N286, "Management System Requirements for Nuclear Facilities" (Reference 14.4-1). The OPG Nuclear Management System sets the standards for health, safety, environment, security, economics, and quality during facility design, construction, and commissioning based on the authority of and a safety culture driven by the OPG Nuclear Safety Policy. The management system promotes a safety culture in which all workers adhere to the management system, implement practices that contribute to excellence in worker performance, supports workers in carrying out their tasks safely and successfully, and monitors to improve the culture.

Pending commercial agreement, OPG's contract model is expected to ensure integration and collaboration with GEH, the constructor, and vendors. As owner, OPG will have the responsibility to ensure ongoing and intrusive oversight for all phases of the facility development following the OPG management system. OPG will require the constructor to have its own management system compliant with applicable current standards. GEH and the constructor will qualify subcontractors to the appropriate quality level, commensurate with the risk that their activities pose to the facility.

The Quality Assurance (QA) Requirements documents align with the GEH internal procedures. The documents are utilized by the GEH team to communicate the OPG and project quality requirements to vendors and subcontractors.

All contractors fully support the QA requirements and implement the policies, philosophies, and practices. Each project professional has access to and is responsible for following the applicable governing documents and supporting program effectiveness through project audits, surveillance activities, and using the deviation and corrective action processes.

A Health, Safety and Environmental Plan is developed jointly by the OPG Health Safety and Environmental Manager and Site Manager, with input from the corporate Health, Safety and Environmental departments of GEH team members. The plan is consistent with the Ontario Occupational Health and Safety regulations and OPG Health, Safety and Environmental policies and programs.

The OPG Operations Program Management Plan details the requirements of the OPG Nuclear Operations Program. However, during BWRX-300 construction, OPG operations and maintenance are not required, thus associated operations and maintenance governance is not required. Commissioning prior to fuel load will be performed under the commissioning lead's (OPG or the constructor) managed system. The OPG Operations Program Management Plan is expected to be implemented prior to the Licence to Operate as a series of standards and procedures that ensure the safety of the public, environment, personnel, and equipment.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

OPG responsibilities cover all aspects related to the construction and commissioning of the facility. These responsibilities include the oversight of contracted activities as well as activities that are specifically performed by OPG to include:

1. Ownership of the safety case (including information provided by design authority, constructors, and contractors)
2. Confirmation that the facility is built in accordance with the design basis, regulatory requirements, and applicable codes and standards
3. Preparation and updating (management) of construction plan documents
4. Performance of inspections, tests, and verification of safety class Structures, Systems, and Components (SSC)
5. Evaluation of safety-significant inspection findings and associated reports to the Canadian Nuclear Safety Commission (CNSC)
6. Providing a point of contact for CNSC communications for all matters related to facility construction
7. Identification of jurisdictional boundaries and responsibilities where more than one regulatory body governs

During Darlington New Nuclear Project (DNNP) preparation, construction and operation phases, Operations team members use a graded approach (where the criteria and process used for grading is defined per the requirements of CSA N286 (Reference 14.4-1)), as applicable to the phase to support the following:

1. Developing and maintaining Operating Experience (OPEX), Risk Management and Significant OPEX Report programs
2. Work Protection
3. Operations Scoping and Assessment
4. Design Package Review and Approval
5. Work Plan Review and Approval
6. Support for commissioning and turnover of plant systems
7. Document Reviews and Governance Development and Revisions
8. Strategic Plans and Schedules Development and Reviews

The processes that comprise the management system and maintain objective evidence to demonstrate effective implementation of the management system are defined, documented, controlled, and maintained.



NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

This section specifically describes the management system and organizational arrangements for the transition from construction to commissioning to operation. The construction and commissioning phases overlap to the extent that the construction phase includes fuel-out equipment checks, referred to as “Check and Test,” and pre-commissioning. Construction, commissioning and related activities are developed using applicable management systems and associated implementing procedures and processes. This section also discusses the programmatic processes that address readiness for operation aspects important to the transition from construction to operation that will include:

1. The organization(s) and organizational structure(s)
2. Measures for assessing the suitability and effectiveness of the plan during all stages of the transition
3. Provisions for recruiting, training, assigning, and retaining the required numbers of workers/ staffing levels consistent with the schedules for implementation and workload
4. Policies, programs, and processes to manage key functions important to safety (i.e., operations, maintenance, and engineering) with a timeline and milestones for development
5. Document management (electronic or paper)
6. Configuration control
7. Transition or transfer of management systems (e.g., construction phase to commissioning phase) to include system and process turnovers
8. The applicability and point at which full implementation is considered complete and OPG assumes management control (in line with the transfer of SSC)

The OPG management system is further described in Chapter 17, Sections 17.1 and 17.2.

### **Organizational Structure and Operation**

The organizational structure implements the programs that make up the OPG management system with the Chief Nuclear Officer accountable for implementation and effectiveness of the nuclear management system. These programs address the following:

1. Organizational resource strategy addressing the quantity of resources and mix of disciplines and skills required as construction progresses through construction and commissioning
2. Description of how resource strategies are managed when project work is being implemented to ensure that the resource profiles and organizational arrangements remain fit for the purpose
3. Description of how contracted work (design, procurement and manufacturing, construction, pre-commissioning checks, and commissioning) is conducted to required levels of safety and quality from an organizational perspective. Considerations to address:
  - a. Evaluation and confirmation before placing a contract with nuclear safety significance that the contractors and suppliers have the organizational, technical and project management capability, capacity, and culture to deliver the items or services
4. The organizational arrangements for transition from construction to commissioning, to operation

### **Management Oversight (Construction/Commissioning)**

Accomplished through OPG's Nuclear Management System in compliance with CSA N286 (Reference 14.4-1) and OPG-PROG-0038, "Contract Management" (Reference 14.4-16), OPG ensures contracted design, procurement, construction, and commissioning work is carried out to the required level of safety and quality by:

- Establishing an effective commercial or supply chain strategy to enable delivery of safety case requirements
- Maintaining an informed (intelligent) customer capability for all work that may affect nuclear safety that is carried out on its behalf by any contractor
- Ensuring the contractor maintains an informed (intelligent) customer capability for all work carried out by the contractor's supply chain that may affect nuclear safety; for example, where a subcontractor may use its own supply chain to meet the needs of its customer, and will need to procure items or services appropriately
- Issuing specifications that adequately describe the items or services, meet the safety case requirements, and identify the required level of QA; some examples are:
  - Procurement specification
  - Commissioning specification
  - Design clarification
  - Codes and standards requirements
  - Description of any operational constraints
  - Review of the specifications and the results of the commissioning activities
  - Disposition and resolution of any design-related performance issues with the SSC, in accordance with a formal design change process for work with nuclear safety significance
  - Prior to contract placement, evaluating, and confirming that contractors and suppliers have the organizational, technical and project management capability, capacity and culture to deliver items or services to the specification
  - Ensuring suppliers have quality management arrangements that are appropriate and consistent with the safety significance of the procured items or services
  - Ensuring suppliers identify and categorize any deviations from specified requirements, and refer the deviations to the design authority and the authority having jurisdiction for assessment
  - Ensuring suitable arrangements to mitigate the risk of Counterfeit, Fraudulent and Suspect Items entering the supply chain
  - Ensuring arrangements are in place to capture and act on operational experience feedback from the safety case and supply chain management activities, sharing learning as appropriate within the organization and wider industry
  - Conducting effective oversight and assurance of the supply chain, including the acceptance of items or services for work with nuclear safety significance

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Consistent with CNSC REGDOC-2.3.1, "Conduct of Licensed Activities: Construction and Commissioning Programs" (Reference 14.4-2), OPG has primary responsibility for the safety and security of construction and commissioning activities, including the work carried out on its behalf by contractors. OPG's oversight responsibilities cover aspects related to facility construction and commissioning.

Additional details of OPG's construction oversight activities and commissioning and turnover plan to address applicable CNSC REGDOC-2.3.1 (Reference 14.4-2) requirements is expected to be submitted in support of the Licence to Construct application via a Construction Management Plan and a Commissioning and Turnover Program Management Plan respectively.

OPG oversees construction of the facility in accordance with OPG's Project Management program. Design oversight is controlled in accordance with OPG's Nuclear Design program and commissioning is consistent with OPG's Nuclear Engineering Change Control program.

The OPG Nuclear Oversight Program periodically assesses the effectiveness of the management system. OPG is expected to provide ongoing and intrusive oversight during the design, construction, and commissioning phases. In addition, OPG requires the construction contractor to also have a management system that is compliant with the applicable standards. The OPG program includes independent audit, self-assessment, and a management review process conducted by OPG senior management. The audit program reviews programs within the management system, including programs that are maintained and implemented as a corporate responsibility. The frequency of audits and selection of program elements to be assessed are based on program risk assessment.

OPG oversight activities include the following:

- Oversight support from corporate Environmental, Safety, Health and Security group
- The Project Quality Manager provides oversight to supplier quality processes, including equipment quality and quality document packages
- The Supplier Quality Surveillance Manager provides oversight of shop inspection activities, inspection personnel for the project and duties of the Quality Control Coordinator
- Construction Management and startup management will have Site Manager and Project Manager oversight

Prior to construction start, interfaces between OPG and the CNSC and other regulatory authorities are defined, agreed upon and understood such that relevant performance issues that affect or have the potential to affect quality of construction and future operational safety of SSC are provided to the regulatory authorities.

NK054-PLAN-01210-00100 Sheet 0003, "Darlington New Nuclear Project Quality Program Management Plan" (Reference 14.4-17), addresses the quality requirements and takes authority from the Program Management Plan. Figure 14.1-1 provides a representation of the links between the OPG Nuclear Management System and the vendor quality systems. See Chapter 17, Section 17.3 for a detailed discussion of OPG quality management.

### **Security (Construction and Commissioning)**

Security measures consistent with the relevant guidance provided in CNSC REGDOC-2.12.1, “High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices” (Reference 14.4-3) and CNSC REGDOC-2.12.3, “Security of Nuclear Substances: Sealed Sources and Category I, II, and III Nuclear Material” (Reference 14.4-4) are established during the construction and commissioning phases for the prevention, detection, and response to, criminal or intentional, unauthorized acts involving or directed at construction or commissioning activities and other intentional acts that could directly or indirectly produce harmful consequences:

- Site physical security
- Personnel security
- Information protection
- Document security
- Cyber security

The DNNP Site Security Plan, NK054-PLAN-61400-00001, “DNNP Site Security Plan” (Reference 14.4-18), defines the processes and procedures that are implemented to control and maintain overall security of the site. Defined roles, responsibilities and content of the security plan are determined by OPG and the constructor. The Site Security Plan is independent of the operating plant security.

The Nuclear Security Program provides for actions to protect SSC and deter conditions that impair site security during the construction and commissioning phases.

The security measures evolve during construction and commissioning commensurate with on-site conditions (e.g., turnover to operations, nuclear material on-site, etc.) with provisions for:

- Commercial loss control
- Access control of personnel, materials, and vehicles
- Scheduled and random patrols and inspections
- Screening (pre-employment and gate clearance) for access to work areas
- Physical barriers, fencing, surveillance and monitoring capability
- Cyber security controls to protect computer-based systems
- Response capability

Revisions to the Nuclear Security Program occur in a phased approach reflecting the stages of the project lifecycle from construction, commissioning, and operations to decommissioning. Nuclear security will increase when nuclear fuel is delivered to the site and further before the reactor is first operated. Measures are implemented appropriate for each phase of the project to ensure compliance with regulation and applicable codes and standards in addition to any measures required to protect personnel, information and physical assets against security risks identified in site-specific threat and risk assessments.

Prescribed and security sensitive information is only provided to persons with a valid security clearance and “need to know.”

Public access to the site-controlled area is restricted by fencing and signage with Security Officers patrolling on a regular basis. Additional security and access control measures are established or modified as appropriate for each phase of the project.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

OPG maintains a security clearance program consistent with CNSC REGDOC-2.12.2, "Security: Site Access Security Clearance" (Reference 14.4-5). Staff and contractors requiring unescorted access to the site require a security clearance commensurate with activities performed or access required. Clearance requirements will increase prior to delivery of nuclear fuel to the site.

Threat risk assessment is performed as part of the Nuclear Security Program with results taken into consideration in plan development and facility response. Agreements with offsite response forces are maintained that provide for an offsite response to OPG facilities. The agreement(s) ensure that necessary resources are available to address design basis security events. OPG periodically conducts drills and exercises that include integrated response with the offsite response force. Lessons learned from drills and exercises are implemented within the security program. Agreements are subject to annual review and are revised as necessary to reflect additional response needed during the various phases of the project.

Physical security measures focus primarily on-site access control with additional equipment, systems, and procedures implemented where required during the construction and commissioning phases.

OPG's Cyber Security Program implements the OPG Cyber Security Policy. Information technology and industrial control systems are managed in a secure, vigilant, and resilient manner that minimizes cyber risks to information assets, renewable generation, and nuclear facilities. The Nuclear Cyber Security Program ensures secure operations of computer systems associated with the industrial control systems for OPG nuclear facilities. Cyber security is applied to plant systems including those used to ensure safe operations and those which provide physical security of OPG nuclear facilities. The Nuclear Cyber Security Program complies with requirements of CSA N290.7, "Cyber Security for Nuclear Facilities" (Reference 14.4-6). Requirements for Cyber Security are documented and coordinated as part of the project contract.

Nuclear Security Officers are selected, trained, and equipped in accordance with the applicable requirements of CNSC REGDOC-2.12.1, "High Security Facilities, Volume I: Nuclear Response Force" (Reference 14.4-40) and CNSC REGDOC-2.2.4, "Fitness for Duty, Volume II; Managing Alcohol and Drug Use" (Reference 14.4-7).

Details (prescribed information) of the security program are transmitted only by secure means consistent with OPG-STD-0030, "Protecting OPG's Information" (Reference 14.4-19) and Section 21-23 of SOR/2000-202, "General Nuclear Safety and Control Regulations", "Prescribed Information" (Reference 14.4-8).

### **Safeguards (Construction and Commissioning)**

Access to the physical site and information about site buildings and structures, operational parameters, flow and storage of nuclear material, and the installation of safeguards surveillance and monitoring equipment is provided to the International Atomic Energy Agency consistent with the Canada-International Atomic Energy Agency safeguards agreement.

OPG has programs in place to facilitate compliance with all applicable safeguard requirements and agreements per CNSC REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy" (Reference 14.4-9). Measures related to site buildings and structures, operational parameters and the flow and storage of nuclear material throughout the lifecycle of the nuclear facility are described in the Environmental Impact Statement.

### **Training (Construction and Commissioning)**

Training programs consistent with CNSC REGDOC-2.2.2, "Personnel Training" (Reference 14.4-10) are established to ensure personnel engaged in construction and commissioning activities are provided with the necessary training and possess the qualifications and competence to perform assigned tasks effectively and safely. NK054-PLAN-01210-00100 Sheet 0007, "Darlington New Nuclear Project Training Program Management Plan" (Reference 14.4-20), under the authority of NK054-PLAN-01210-00008, "Darlington New Nuclear Project - Program Management Plan" (Reference 14.4-21), addresses the development and delivery of training for OPG staff and contract staff. The training programs are expected to evolve as construction progresses and the commissioning and operational phases are entered. Training is often integrated within specific programs or plans with links established and maintained with established training programs.

Vendors employing staff, supervisors, contractors, and subcontractors who are working on the project are responsible for the training and qualification of their staff in all training topic areas under the vendor's QA programs. OPG provides vendors with site-specific information where needed. Skilled Trades staff hold journeyman status and Certificate of Qualification as appropriate. Apprentices work under the accountability of a journeyman when performing tasks associated with the skilled trade.

Vendors maintain records of staff certifications, licences, training, and qualification and are available for OPG review on request. OPG Security personnel are trained under N-PROG-TR-0005, "Training" (Reference 14.4-22) including N-TQD-603-00001, "Nuclear Security Training and Qualification Description" (Reference 14.4-23).

Scheduling of OPG staff training for DNNP follow existing processes per N-PROC-TR-0044, "Training Demand, Scheduling, and Cancellation Process" (Reference 14.4-24) and align with staffing plans for the new station. Processing of results and maintenance of training records follow existing processes described in N-PROC-TR-0041, "TIMS II Administration" (Reference 14.4-25) and N-PROC-TR-0012, "Records and Documentation" (Reference 14.4-26).

Personnel engaged in construction activities are provided the training necessary to perform their assigned tasks effectively and safely. The training content is specific to the tasks performed and emphasize adherence to established programs, processes, and procedures to assure nuclear safety and, every person working at the site is responsible for safety.

The respective training for personnel engaged in commissioning activities to include:

- Commissioning organization and structure
- Commissioning procedures
- Reactor facility systems
- Conduct of testing and maintaining safe conditions
- Procedural and design changes
- Design process as it applies to configuration control and field changes
- Permanent and temporary modifications
- Work control and equipment isolation
- Interfaces of construction, design, and operation with commissioning

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

- Test limitation boundaries in mechanical, instrumentation and control and electrical systems
- Reporting incidents and deviations
- Commissioning methods and techniques
- Safety culture
- Nuclear safety, industrial safety, fire protection, emergency preparedness, radiation protection and security
- Industry OPEX and lessons learned
- Design criteria, technology and operational limits and conditions (or the equivalent) for the reactor facility
- Environmental protection and waste management of spent fuel and radioactive waste
- Full-scope simulator training of operators for reactor startup, regular operations, reactor shutdown and cool down and handling of various transients, including accidents

Training requirements will be implemented and maintained and will be consistent with the requirements pursuant to the phase of the project. When required, site personnel are initially provided general Site Orientation training which consists of topics that include fall prevention, confined space, aerial lifts, hazardous communication, ladder safety, excavation, and trenching. Additional training for specialized tasks is addressed on an individual need basis.

Documentation of employee certifications and education is recorded upon hiring by either OPG or vendors and contractors. Competencies, skills, and acquired knowledge is continuously monitored and documented.

Short term visitors to the site are provided with an orientation and overview of current site hazards. Visitors are required to provide written acknowledgment of the guidelines to be followed. Escorting of visitors will be used when conditions warrant additional oversight.

Minimum Staffing Complement is discussed in Chapter 13, Subsection 13.2.2.

### **Programs and Procedures (Construction and Commissioning)**

OPG's Nuclear Management System programs and a description of the specific programs associated with the safety culture elements of OPG's Management System are provided in Chapter 17, Subsections 17.1, 17.2 and 17.5 respectively.

Operating procedures developed for use during construction, installation and commissioning are developed under the same general guidelines as described for operating procedures developed for licensed operation of the facility with the exception that the primary focus of these operating procedures is directed to an evolving facility as it is constructed. Construction and commissioning phase operating procedures are established to provide for the safe conduct of operations and evolutions performed during these phases. The construction phase and commissioning phase procedures are primarily developed under procedure guidelines that are consistent with the requirements of CNSC REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility" (Reference 14.4-11).

The development, review, and approval of applicable procedures and programs, including necessary changes after approval, are governed by N-PROG-AS-0001, "Nuclear Management System Administration Program" (Reference 14.4-27), and implemented in accordance with OPG-PROG-0001, "Information Management" (Reference 14.4-28).

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

The schedule for development of plant operating procedures is based on completion of finalized design of SSC and later the complete plant site. Operating procedures are projected to be developed approximately 1 year prior to the start of pre-operational testing of systems. Plant operating procedures are used and tested to the extent practical during the pre-operational and startup testing. Technical procedures are developed under N-PROC-AS-0028, "Development, Review and Approval of Technical Procedures" (Reference 14.4-29). The plant simulator is expected to be used to validate specific operating scenarios.

Processes are established for the development and approval of test procedures that control the performance of tests and the review and approval of test results including the required actions to be taken when test results do not meet design requirements.

Processes are established for the development and approval of procedures for receipt inspection of fuel, fuel handling, fuel storage, initial fuel loading and initial criticality that include the protection and safety measures established for safe operation. Fuel receipt, storage, and initial core loading is controlled via procedure, which contains specific instructions and guidance to receive, handle, store, and load the fuel safely, correctly, and efficiently.

Out of core criticality is precluded by fuel storage design and the storage and handling procedure guidance.

Document and records program development and management during construction, commissioning and operation is the responsibility of the operating organization. Construction and commissioning phase documents and records are maintained during the applicable phase by the responsible organization (GEH, constructor, etc.) in an agreed upon form to facilitate their turnover to OPG. Document and records management is discussed further in Chapter 13, Subsection 13.3.8 and Chapter 17, Sections 17.2 to 17.4.

### **Operating Experience and Problem Identification (Construction and Commissioning)**

Relevant OPEX is considered for the BWRX-300 during the construction and commissioning phases. The design authority (GEH) establishes provisions for the incorporation of OPEX through integrated management systems. The OPEX comes from a variety of sources including direct input, GEH/GNF experience from the operating Boiling Water Reactor and Advanced Boiling Water Reactor fleet, Institute of Nuclear Power Operations, Electric Power Research Institute, Department of Energy, U.S. Nuclear Regulatory Commission, and CNSC. OPEX is also considered for use in test programs where special emphasis might be warranted. The constructor OPEX program is consistent with the OPG OPEX program and is implemented accordingly during construction planning and construction. OPG's problem identification and resolution process implements a program to take corrective action from facility events and has in place an established operating experience process that evaluates, integrates, accesses and shares OPEX information to prevent event recurrence and to initiate improvements. The problem identification and resolution program and OPEX programs are maintained consistent with CSA N286 (Reference 14.4-1). Problem identification and resolution requires that when problems arise; they are immediately controlled and documented if required, evaluated for significance and the underlying cause if deemed to be systemic or having an impact on business objectives, and accepted. The OPEX program for the operations phase is addressed in Chapter 13, Subsection 13.3.7.

Processes and programs are established for problem identification, resolution, and continual improvement during commissioning with expectations set for personnel to identify and report non-conformances. The GEH and construction problem and identification programs are consistent with the OPG program.



Based on construction lessons learned, measures are taken to mitigate the effects of adverse weather by construction means and methods addressed in the Construction Plan. Severe weather that may impact construction is also considered with measures to mitigate the impact and thus limit, but not prevent, effects to the Construction Program.

### **Nuclear Material Packaging and Transport (Construction and Commissioning)**

A Nuclear Material Packaging and Transport program, specifications, and procedures, established by the constructor, will be in place to cover all requirements for the construction and commissioning phases of the project. OPG/GEH is expected to ensure this constructor program meets CSA N286 (Reference 14.4-1) requirements.

Further details on the Nuclear Materials Packaging and Transport Program are provided in Chapter 13, Subsection 13.3.2.

This section does not include detailed information pertaining to operations. The information provided is intended to facilitate pre-commissioning checks and commissioning before fuel load and prepare for the transition to fuel-in commissioning and reactor operation upon receipt of a Licence to Operate.

### **Emergency Management and Fire Protection (Construction and Commissioning)**

Emergency management and Fire Protection for DNNP will adhere to the requirements of CNSC REGDOC-2.3.1 (Reference 14.4-2). All contract workers are required to complete mandatory emergency response training that outlines rules, notifications and required responses.

OPG developed the DNNP Nuclear Emergency Preparedness Plan. The plan provides a basis to document the concepts, roles, and resources required by OPG to implement and maintain its emergency response on the DNNP site to protect employees, visitors, and contractors in the event of a nuclear emergency originating from Darlington Nuclear Generating Station operations. See Chapter 19, Section 19.1 for further details.

Nuclear emergency response is based on the existing Darlington Nuclear Generating Station nuclear response requirements and CNSC REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response" (Reference 14.4-12).

Emergency measures are routinely evaluated to ensure that they remain commensurate with on-site hazards.

#### **14.1.1 Configuration Control**

The design and safety analysis are incorporated into the purchasing, construction, commissioning, operating, and maintenance documentation such that the as-built configuration of the facility is aligned with the design and safety analysis. Design authority configuration requirements that include the responsibilities and authority of organizations whose functions affect the configuration of the facility, including activities such as design, maintenance, construction, licensing, and procurement are controlled through the configuration management process. OPG configuration management is an integrated management process that ensures the physical and operational configuration and documentation continue to conform to the design and licensing basis requirements. Configuration management during the construction phase is implemented through processes (design changes, field changes, etc.) agreed upon by GEH, constructor and OPG in consideration of the as-built documents to be turned over to OPG.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Facility configuration is maintained from initial conception through established programmatic configuration and change control processes that adhere to CNSC REGDOC-2.3.1, Section 7 (Reference 14.4-2), CSA N286.10, "Configuration Management for High Energy Facilities" (Reference 14.4-13) and CSA N286 (Reference 14.4-1). The control of construction records is established at the beginning of the construction phase for schedule input.

Changes during the construction phase are processed to maintain conformance with design requirements, physical configuration, and configuration documentation with established arrangements between participating organizations for review, approval, and release, including notifications of field changes and non-conformance issues. Changes during the construction and commissioning phases are managed by the design authority with oversight and concurrence of OPG.

The design control process ensures engineering documents, calculations and detailed design drawings are generated and the corresponding GEH team, engineering service, or constructor procedure requirements for review, verification, and approval are completed. The design control procedures ensure appropriate reviews of manufacturers' drawings and data to ensure that the OPG requirements are incorporated in the plant design and documentation.

The construction contractor maintains detailed design control and engineering change notification procedures specific to their individual organization or implements procedures belonging to the organization responsible for the design. Selected construction contractors are required to demonstrate how their specific design control and engineering change procedures are compliant with CSA N286.10 (Reference 14.4-13).

Changes to the approved design are controlled, reviewed, and approved in the same manner as design review and verification. An engineering change notification, or equivalent, is used to provide interim notification of a change to be incorporated. Engineering Change Notifications are processed in accordance with the design change management process.

GEH also maintains an issue tracking system for change management during construction. Changes that may be the responsibility of OPG are submitted to OPG in accordance with the project contract.

#### **14.1.1.1 Facility Configuration**

Configuration management is incorporated into all aspects of purchasing, construction, and commissioning so the as-built configuration of the facility aligns with the design and safety analysis in accordance with CSA N286 (Reference 14.4-1) and CSA N286.10 (Reference 14.4-13).

Configuration information that describes, specifies, certifies SSC, or provides data or results created during construction are agreed to, planned, and processed to facilitate turnover for commissioning and operations. Control of construction records are established prior to beginning the Construction Program as part of construction schedule input. Visual documentation of the as-built condition is appropriate as part of configuration documentation, particularly in inaccessible areas or areas subject to radiation exposure.

#### **14.1.1.2 Change to Facility Configuration Information**

The design basis and requirements for the BWRX-300 including safety analysis is established and documented in accordance with CSA N286 (Reference 14.4-1) and is traceable to the respective SSC. Changes/modifications to the facility configuration during construction are processed and documented to maintain the facility design requirements, physical configuration, and the configuration information. Impacts of design changes are assessed, addressed and when applicable reflected in the safety analysis. Relevant configuration changes will be communicated to the CNSC in accordance with applicable licence conditions.

Temporary or permanent changes in design requirements, physical configuration and physical configuration information are configuration change mechanisms that include:

- Design changes
- Field changes
- Non-conformances
- Changes to as-built condition
- Changes to as-built test documentation
- Changes to inaugural inspection records
- Computer software changes
- Changes to records of maintenance history
- Temporary modifications and alterations

#### **Fitness for Service**

During the design and construction/procurement phases, equipment reliability requirements are addressed within the design authority's reliability program. During the startup and testing phase when transitioning from construction/commissioning to operation, equipment reliability requirements are addressed as a shared responsibility of the design authority (GEH) and OPG. The Equipment Reliability Programs of both organizations are consistent with CNSC REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants" (Reference 14.4-14).

The fitness for service safety and control area covers activities that affect the physical condition of SSC to ensure adequacy and ability to perform their intended function when required.

The Chemical Control Program establishes processes to control the use of chemicals throughout the facility during construction, commissioning, and operations.

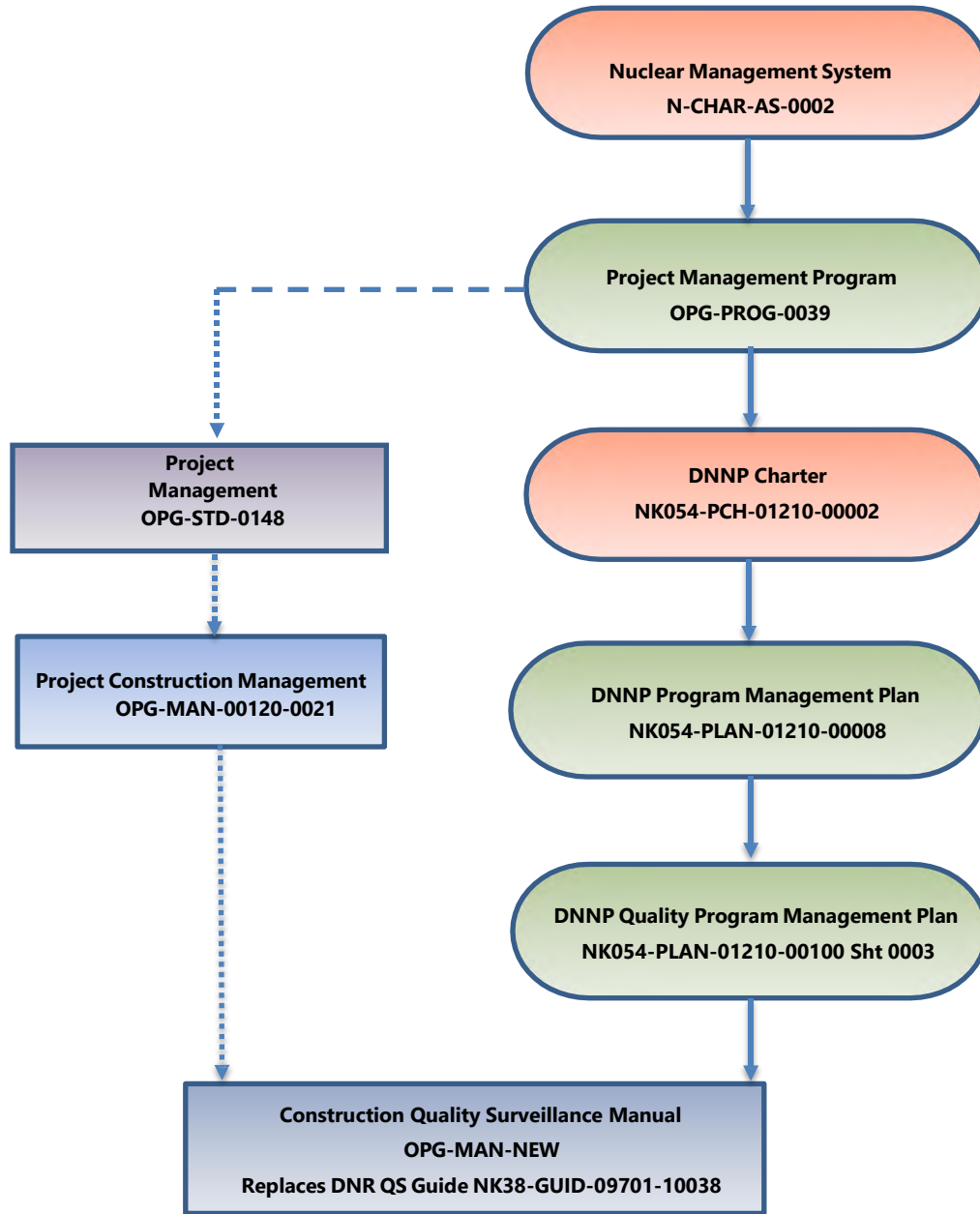


Figure 14.1-1: Quality Program Process Documents

## **14.2 Construction**

### **14.2.1 Organization**

#### **OPG Construction Team Organization**

The Construction Program Management Plan, NK054-PLAN-01210-00100 Sht. 00009, "Darlington New Nuclear Project Construction Program Management Plan" (Reference 14.4-30), provides guidance as well as the activities for the OPG construction oversight team engaged in the planning, construction, and commissioning of the BWRX-300. The Construction Program Management Plan aligns with Corporate, Nuclear, and business unit governance.

The construction organization performs independent risk-based observations focused primarily on High-Risk areas. This allows for proactive identification of areas for improvement that can be actioned to have corrective measures implemented.

Construction Management is the responsibility of the individual engineering service or Construction contractors. OPG Construction Managers, reporting to the OPG Director of Field Construction, support the Project Managers/Directors proactively with qualified and competent resources that have the experience required to execute effective oversight management. Effective communication both vertically and horizontally between the construction organization and the Project ensures alignment for field construction activities. Construction oversight is a project management function and is accomplished through OPG-PROG-0039, "Project Management" (Reference 14.4-31).

The typical OPG Construction Team organization consists of a Director of Construction, Construction Manager, Construction Supervisor, and Quality Officer.

Roles and accountabilities of the OPG Construction Team organization are described below:

#### **1. Project Director Construction Services**

The Director Construction Services is accountable for the oversight of the construction activities of the new nuclear plant. The Director interfaces with the General Contractor and Subcontractors to ensure project deliverables are met with due consideration for the site-specific safety and environmental regulations and requirements. The Director provides oversight of the General Contractor's compliance with the reviewed design and construction documents and the construction and installation methodology. The Director also provides oversight of the General Contractor's QA/Quality Control Program to ensure that regulatory and contractual requirements are met and that the work is performed by qualified staff.

The Director Construction Services also provides direction to the OPG Field Construction Managers on contractor observation in line with OPG-PROG-0039 (Reference 14.4-31) and ensures the Construction Manager is providing independent risk review of Major Projects to ensure vendors are ready to execute the work when scheduled.

2. Construction Manager

The Construction Manager reports to the Director of Construction Services and provides a point of accountability for implementation of the construction project, ensuring that all aspects of the DNNP work program are executed in accordance with expectations for safety, cost, schedule, and quality.

The Construction Manager provides leadership, strategic direction, support to assigned Construction planning and execution personnel and ensures that the engineering service vendor, and other vendors and subcontractors, deployed on the project executes the work to the schedule and consistent with the terms and conditions of the contract. This manager also ensures that the engineering service vendor performs work safely, effectively and in a manner consistent with OPG's policies, procedures, safety values and objectives.

3. Construction Supervisor

The Construction Supervisor reports to the Construction Manager and is required to support early design reviews at the onset of detailed design. Planning personnel are co-located with the engineering service vendor team to engage collaboratively, be efficient and effective in the design review and planning phases, assist with the development of the project plans, and support the engineering service construction execution planning efforts.

4. Quality Officer

The Quality Officer has the following responsibilities:

- a. Prepares Quality Surveillance Plans that inform the assessment of vendor compliance to quality programs, plans, and processes
- b. Conducts specified surveillance and records observations in an electronic observation repository
- c. Reports metrics and trends based on the observations and identifies adverse quality non-conformance trends and Significant Quality Events to the Senior Manager Assurance and the appropriate Project Manager
- d. Recommends strategies and corrective actions for quality issue resolution
- e. Ensures configuration management by surveillance of documents
- f. Prepares Quality Final reports for each major project

**Consortium Construction Team Organization**

The planned project structure that is being defined in the Project Contract Model (Chapter 17, Subsection 17.2.1.1) will result in a consortium or alliance involving OPG (owner), GEH (developer), a constructor and an architect engineer.

Subject to the final terms of the model, at the consortium level, key decision-making and high-level project leadership will be undertaken by a Project Leadership Team and an Executive Leadership Team. The team will consist of a Project Director from each member organization.

For construction leadership, the constructor will be under the overall direction and leadership of a Project Director. Project Managers and potentially reporting Directors will be accountable to the Project Director.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Construction Leadership will roll down from the Project Managers, who are accountable for specific scopes or bundles, to the Construction Team, led by a Construction Manager and General Superintendent. The hierarchy will cascade down through discipline superintendents, General Forepersons, Forepersons, and ultimately tradespersons.

The construction field staff organization consists of a project site manager, Project Manager, general superintendent (Construction Manager), trade discipline superintendent, project controls lead, technical services lead, quality lead, health, safety and environmental lead, field office lead, general office administration, subcontracts coordinator, materials lead, and trade general foremen.

1. The project site manager is the senior site leader responsible for total project site construction performance including safety, costs, schedule, quality, and project status. The project site manager oversees operations of project personnel and maintains relationships with clients, engineers, and subcontractors.
2. The Project Manager is responsible for project performance including safety, costs, schedule, and project status. The Project Manager assists the project site manager with project management, project controls and reporting and acts in his/her place during the project site manager's turn around rotations.
3. The general superintendent(s) (Construction Manager), who manages all field construction on the project and is responsible for all field aspects of the project's budget, schedule, safety, resources and general performance, reports to the project site manager.
4. The trade discipline superintendents report directly to the general superintendent(s).
5. The project controls lead (or manager) reports to the project site manager. The project controls lead has overall responsibility to ensure that all project controls systems as related to cost control, progress monitoring, planning, and scheduling, and document control are implemented and delivered successfully.
6. The technical services lead, whose responsibilities include workforce planning, job site coordination, translating the interpretation of drawings, and specification into work instructions, reports to the project site manager. The technical services lead assists the general superintendent and project site manager in the management and coordination of the technical and administrative requirements of a project.
7. The quality lead reports to the project site manager and supervises the quality inspectors assigned to the project.
8. The Health, Safety and Environmental lead reports to the district Safety Manager and project Site Manager. This lead works closely with the OPG Construction Manager and provides the necessary expertise to ensure the construction effort complies with all GEH, or constructor, OPG, federal, provincial, and local jurisdictional authorities and project safety and loss prevention requirements.
9. The field office lead, reports to the project site manager and provides accounting, payroll, and project administrative expertise to support the construction and accounting teams for the project.
10. The general office administration is responsible for project office security, project office policies and procedures, systems and technology requirements for the project, and office equipment, office supplies, and furniture needs.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

11. The subcontracts coordinator formulates, negotiates, and executes subcontracts in accordance with the procurement/materials management process and the requirements for subcontract close-out.
12. The materials lead (or manager) develops and implements the project materials management system in alignment with the Materials Management Plan and looks after the development of project materials coordinators.
13. The trade forepersons report to a trade general foreperson(s) and trade superintendent and are responsible for directing and leading crews of hourly workers (trades) including all aspects of the crew's work including safety, quality, planning, production, look-ahead and measurement of performance (typically earned value and schedule adherence).

Terms and titles used above related to staffing positions may be revised during the project life cycle. Some roles will be combined during project ramp-up and ramp-down. The titles and roles listed represent the core construction functions to be performed by the construction organization.

The Construction Manager draws on the support available from within the full GEH team for advice and assistance, particularly when approaching new situations or when issues arise during implementation.

The construction management staffing schedule defines the GEH field team personnel by title, forecasted dispatch and departure dates, and location of work. De-staffing/demobilizing is forecasted within the staffing plan.

Trade labour for the project consists of a significant number of employees from the engineering service/constructor firm(s) and their affiliated Trade Unions, their subcontractors, and suppliers, and OPG that are local residents with the intent to maximize the amount of local content in the local community and province of Ontario. Applicable labour laws, including provincial and federal regulations are adhered to.

Prime, partner and subcontractor roles for construction are identified in the Construction Plan along with the identification of construction contracts that may start out in or remain under OPG control. The roles are subject to change as construction progresses. GEH is responsible for the majority of the small modular reactor engineering and related procurement while the engineering service/constructor manages the majority of on-site work and procurement of balance of plant materials.

During construction, the construction organization installs and erects plant equipment and performs construction and installation testing, typically via inspection and test plans. As construction and installation testing is completed and construction completions are declared, equipment and systems are ready for the execution of the pre-operational and startup testing by the licensee's operating organization. To ensure a smooth transition from the construction organization to the operating organization the licensee will have a commissioning organization.

A construction to procurement interface, described in the Construction Plan, is established to ensure that procurement requirements adequately capture construction plan input for skids, modules, shipping condition, instruction manuals, etc. The engineering service has established interface arrangements with the procurement team, GEH, key suppliers (e.g., steam turbine generator), and key subcontractors to develop construction plan input and convey it into procurement requirements.



## **Construction Management**

OPG is responsible for the identification of the health and safety, environmental, and other requirements applicable to construction activities and the communication of relevant requirements to all parties. The relevant requirements are taken into account and implemented via the established OPG management practices and controls.

The construction organization, in general, consists of the design authority, the engineering service, constructor, and various equipment vendors and subcontractors performing construction activities under OPG oversight.

The plan for the project setup, identifying OPG as "owner only" during construction, will be confirmed by the Contractor Owner Interface document which is currently in progress. OPG is highly involved with project execution from the start of the project through the end of the warranty period. OPG is expected to have multiple team members on-site during project execution.

Interface arrangements between OPG, the CNSC and other regulatory authorities are established and documented in the DNNP Licensing Program Management Plan NK054-PLAN-01210-00100 Sheet 00008, "Darlington New Nuclear Project - Licensing Program Management Plan" (Reference 14.4-32).

The satisfaction of contractor and subcontractor contractual obligations are ensured via processes and procedures developed in conformance with the OPG management system. OPG maintains oversight activity records and provides reports to the CNSC of any relevant contractor performance that has affected, or has the potential to affect, the quality of construction and future operational safety.

OPG oversight of contractor activities is addressed in the Construction Project Assurance Program Management Plans and includes:

- Qualitative and quantitative measures to monitor conformance and trending
- Proactive measures that monitor contractor performance
- Reactive measures to trend contractor performance
- Monitoring management system/QA program effectiveness
- Retention of relevant information and reports of contractor performance that has affected, or potentially affected, the quality of construction and future operational safety

The OPG Construction Organization facilitates construction activities in line with OPG-PROG-0039 (Reference 14.4-31), with the details provided in OPG-STD-0148, "Project Management" (Reference 14.4-33), OPG-MAN-00120-0021, "Project Construction Management" (Reference 14.4-34) and N-INS-00120-10008, "Nuclear Contractor Safety Management Process" (Reference 14.4-35). OPG Field Construction conducts activities to create alignment, as well as cover accountabilities and legal requirements under the Occupational Health and Safety Act as supervisors of the work when OPG is the constructor. This alignment ensures a strong relationship is maintained between the Project and vendors while work is being performed in the field.

The OPG Construction Organization performs both scheduled and emergent independent risk-based field observations of the vendor's conduct in specific field focused activities.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Look-Ahead Teams review Project readiness including Comprehensive Work Packages, Task instructions, foreign material exclusion planning, inspection and test plans, Work Plans, assessing details including tasks for all activities, sequencing, task durations, Materials, Safe Work Plans, Lift Plans/Lift Classification, Approvals, Work Perimetry, Resources, Field Walk downs, Permits and Potential schedule impacts.

Work stoppage events pose significant risks to the project cost and schedule. To mitigate these risks, strategies are implemented for event management and event recovery. An event reporting protocol is provided to aid in corrective action(s) and development of recovery plans to reduce the impacts to the project and the site.

Weekly progress reports for construction and monthly progress reports for the project that summarize the performance and status of the project are expected to be issued. The specific content of the reports is expected to be outlined in the project contract.

A Project Controls calendar is used to define detailed dates for status reports issued throughout the project.

Periodic meetings are scheduled and conducted by the Startup Manager that consist of a review/status update of the overall schedule, system turnovers, punch lists, startup progress, startup activities for the upcoming work period, and operator training programs.

The monthly progress report summarizing the performance and status of the project is issued to OPG prior to the monthly meeting. The meeting attendees include the Project Manager, Assistant Project Manager, Project Field Manager, Original Equipment Manufacturer Startup Managers, and other applicable key personnel as may be required based on project phase.

OPG and the GEH team conduct monthly meetings for the purpose of reviewing the work scope progress, progress reports, health plan, environmental plan, safety plan, quality program, and adherence to the project schedule.

The Initial Test Program consists of a series of tests categorized as Construction Testing ("Test and Check"), Pre-operational Testing, and Startup Testing. A startup planning meeting for construction testing, pre-operational testing, and startup testing is expected to be performed.

A performance test meeting is conducted at least 120 days before commencement of the first performance test to finalize the initial coordination of the various tests. The meeting attendees include the Project Manager, Assistant Project Manager, Project Field Manager, Original Equipment Manufacturer Startup Manager, and other key personnel.

Engagement of the GEH team, OPG, and key subcontractors and vendors regularly occurs in the form of project execution planning workshops throughout the project life cycle.

The OPG oversight requirements are extended to contractor obligations to ensure its subcontractors meet their respective obligations.

Constructability workshops are held to finalize coordination between early site works subcontracted by OPG and construction. The workshops are continued throughout the project life cycle as detailed information becomes available.

A lesson learned/OPEX process is utilized through all phases of project execution. The constructability program addresses the lessons learned and implements as applicable. During construction, the team documents significant engineering, process, or construction lessons learned as they occur or, at a minimum, monthly. The appropriate lessons learned are collected and documented in the project files.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

The constructor has an approved American Society of Mechanical Engineers (ASME) Quality Program for design, fabrication, construction, and assembly of materials, systems, components, and parts governed by the ASME Boiler and Pressure Vessel Codes and National Board Inspection Code.

The constructor has appropriate ASME/National Board Inspection Code stamps and certificates necessary for the work to be performed to include S, A, U and PP, U2 & R Certificate Holder to design, fabricate, and assemble power boilers, pressure vessels, and power piping which conform to the requirements of ASME Boiler and Pressure Vessel Code, Section I, "Power Boilers", Section VIII, Division 1, "Pressure Vessels," Section VIII, Division 2, "Pressure Vessels, Alternative Rules" and ASME B31.1, "Power Piping" and the ability to perform alterations and repairs to the National Board Inspection Code and per applicable jurisdictional requirements.

For nuclear work, the constructor has appropriate ASME/National Board Inspection Code stamps and certificates necessary for the work to be performed to include N, NA, NPT, NS, CC and NR Certificate Holder to design, fabricate, and assemble components, containment, and piping which conform to the requirements of ASME Boiler and Pressure Vessel Code, Section III Division 1 and ASME Section III Division 2 and the ability to perform alterations and repairs to the National Board Inspection Code and per applicable jurisdictional requirements.

For nuclear work, the engineering service/constructor performing the work has the ASME N, NA, NPT, and NS stamps applicable to the work performed.

All members of the project management and supervision team are required to be familiar with project specifications and the contract (through self-reading or training). Construction inspection and test plans are based on design requirements and project specifications, contract requirements, local, state/provincial, and national code requirements, and specific vendor or manufacturer recommendations. These documents are distributed and discussed with the project team for a clear communication and understanding of all requirements, roles and responsibilities, and inspection points (Hold, Witness, Review). These documents are communicated to the workforce by use of installation work packages to ensure that the elements of the field quality control are applied.

Prior to the start of construction, quality requirements are identified for each work activity and planned into the work to invoke necessary in-process verification steps and controls for quality and to ensure compliance with the contract and other associated requirements. A "Work Plan" process is utilized to select methods of construction, verify material, review safety and quality concerns, and provide required quality documentation. The following is an example of some of the elements reflected in work packages with respect to quality

- Construction Inspection Test Plans
- Inspection and Test Records
- Acceptance Criteria
- Specification Tolerances

Where deviations are anticipated or required, they are managed and approved in accordance with project team approved manuals or procedures and stored in project files. If the management team has determined that it is necessary to deviate from governing documents, a formal request for deviation must be approved by the process owner before an alternate process occurs.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Whenever non-conformances in materials, systems, components, parts, services, or workmanship are identified, assigned quality personnel (e.g., Project Quality Manager, Site Quality Manager, or Site Engineer) are notified. Non-conformances are documented, tracked, with disposition determined in accordance with approved procedures and stored in project files.

The GEH corrective action program is used for identifying improvement opportunities and other conditions adverse to quality. Corrective actions are focused on eliminating causes of the nonconformities to avoid recurrence. Refer to the approved GEH team quality manuals or similar manuals for Construction.

GEH formal audits are performed during construction to ensure the following processes and activities are effectively implemented in accordance with the governing requirements:

- Project Management
- General Construction and Site Infrastructure
- Document and Project Control
- Requisitions and Subcontracts
- Material Control
- Quality Control
- Welding and Non-Destructive Examination
- Civil Works
- Structural Steel Erection
- Boiler and Pressure Vessel Assembly
- Equipment Installation (Static and Rotating)
- Piping and Pipe Support Fabrication and Installation
- Electrical Construction
- Instrumentation and Controls
- System Turnover and Completion

During construction, official project correspondence, memoranda, calculations, drawings, procurement specifications, design specifications, stress reports, test reports, manufacturers' material certifications, manufacturers' drawings, and similar project records are considered controlled project documents. Project and quality records are controlled in accordance with quality management system requirements. Sensitive and safeguards information receive special handling as determined by GEH, OPG, Canadian, and United States requirements (as applicable).

Site documentation is maintained on-site and in the document management system. Document control administration (e.g., managing inflow and filing of documents) may be managed from the corporate (home) office. Site access to document files is controlled through the document management system.

Drawings are issued in accordance with the agreed-to project policies and procedures. Drawings are transmitted electronically, and hard copies are produced at the project site only on an as needed basis. Hard copy drawings are controlled in accordance with the project quality requirements.

Shop drawings are reviewed by the engineering and Construction Team before being released for fabrication.

Issued construction drawings are managed by the field engineering manager using the same system used by Engineering, including the revision process.

An audit process is implemented by the field whereby the drawings that are in the field are reviewed for correct revisions. The process is managed electronically, including workflow process verifications of the correct version being used.

Work packages without as-built redlines are not accepted as complete. The redline drawings are maintained in the document management system and copied to engineering. The drawings are retrievable from the document management system. As-built drawings associated with ASME Class 1, 2, or 3 piping and associated supports are processed into an engineering workflow for ASME Code Reconciliation.

#### **14.2.2 Existing Facility Effects**

The effects of hazards to or from near site facilities are considered in the assessment of safety and security during construction. The consequences of potential contamination (nuclear or hazardous substances) from a construction site to operating units and from an operating site to the construction sites are considered and assessed in Chapter 2, Sections 2.2, 2.4 and 2.8. The consideration includes an impact assessment of the cumulative environmental discharges of all facilities on the site.

Emergency response planning and response is discussed in Chapter 19.

The responsibilities of relevant licensees and construction organizations for safety and security are agreed upon before the start of construction activities with close communication established between the parties.

The boundaries of physical, system, controlled areas, security access and clean zones are identified for adjacent installations or with common buildings or services. If existing nuclear installation resources are used (e.g., water, electric power, or security), clear interfaces and limitations are defined so that the operating units and related facilities are not jeopardized.

Procedures are established that require an endorsement of a change of status for common buildings or services before construction work plans are put in place.

#### **14.2.3 Construction Readiness Review**

Prior to the beginning of construction, the readiness of contractors to proceed is ensured by a construction readiness review consistent with CNSC REGDOC-2.3.1 (Reference 14.4-2). The construction readiness review verifies:

- Management systems are in place
- Adequate planning has been conducted
- Procedures and training are completed as necessary
- Construction hazards have been adequately evaluated with control measures identified
- Environmental controls are in place consistent with assessed risks and potential or planned impacts

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

The construction readiness review assesses the following areas:

- Regulatory requirements
  - Satisfaction of all applicable regulatory requirements and all required permits (federal, provincial, and municipal) obtained.
- Management system
  - Key construction positions are established with related organizational roles and responsibilities known with the project sufficiently staffed to oversee construction.
  - Management systems are in place to monitor performance against the project baseline.
- Design completion
  - Design is sufficiently complete to allow the construction readiness review verification steps listed above to be undertaken. Incomplete areas are identified, and schedules established for completion.
- Information technology
  - Alignment and interoperability of hardware, software, information communications and the information technology environment for communications with contractors.
- Construction procedures
  - Contractor and subcontractor procedures used for completion of the facility construction in accordance with applicable regulations, design, and contract requirements.
- Materials management
  - Process for construction activities, including the acquisition of materials, delivery, inspection, packaging, storage requirements and waste management from materials receipt.
- Health, safety and environmental assurance
  - Capability of the constructor to manage a safe project that includes safety management system key requirements, specific plans and procedures related to industrial health and safety, industrial hygiene, and environmental controls. Verification that contractors have a completed project safety and health plan and environmental management plan.
- Project control
  - Adequacy of project controls that ensure adherence to the performance baseline and the systems or processes relied on for monitoring and controlling the project.
- Construction execution plan
  - Specific construction activities and the qualified personnel and procedures in place to accomplish the work; to include general construction topics such as site preparedness and work sequencing.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

- Training and qualification
  - Training and qualification of personnel responsible for construction activities to encompass the general training required for site access and specific training necessary to perform planned work activities.
- Work planning
  - Work processes are controlled by approved instructions, procedures, design documents, technical standards, or hazard controls appropriate for the task performed.
  - The organization of work and whether systems are in place and sufficiently mature to support the development of work packages or processes.
- Constructability
  - Design specifications, drawings, site conditions and construction schedule are reviewed by the construction organization and deemed practicable and efficient.
- Field engineering
  - Readiness explicit to construction of specific facility systems in accordance with the approved design, taking into account field observation feedback that may impact design.
  - Field staff in place to support construction with technical guidance and oversight and ensure adherence to the design requirements.
- Infrastructure
  - Support systems including required electricity, gas and water supply, fire protection, temporary offices and sanitation facilities, protection of SSC after installation (including environmental qualification requirements).
- QA
  - Verification of an approved QA plan to address construction and procurement activities.
- Labour management
  - Labour management necessary to successfully execute the project and ensure the adequacy of the local labour force to support the project.
- Construction tools and equipment
  - Availability and operability of tools and equipment necessary to support construction activities and ensuring the equipment meets jurisdictional requirements.

Construction of SSC is established and controlled using generally accepted construction and project management practices in accordance with the design documents. Construction activities are controlled in accordance with design drawings, specifications, and procedures that include:

- Prerequisites
- Precautions to be observed
- Installation requirements
- Sequential actions to be followed, including coordinating construction and verification activities

- Inspection and test plans
- Special equipment and procedures required for installation
- Specific document reference
- Data report forms and records
- Reviews and approvals
- Housekeeping requirements
- Foreign material exclusion requirements

#### **14.2.4 Construction Program**

The Construction Program, accepted by OPG, establishes the planning, scheduling and construction sequencing. Hold and witness points are identified with provisions for interested parties such as engineers, architects, inspectors and CNSC staff. Right of access to facilities and records for witness points or audit by the CNSC is assured.

Items with long lead times, on-site manufacturing, modular assembly, and testing are identified with provisions to ensure construction sequencing is not adversely affected. Any differences between purchasing requirements, the licence to construct design basis and as-built items are evaluated, reconciled, and reported to the authorized inspection agencies and the CNSC. Long lead items for the BWRX-300 construction include, but are not limited to:

- Reactor Pressure Vessel
- Hydraulic Control Units
- Fine Motion Control Rod Drives
- Steam Turbine Generator Set
- Reactor Pressure Vessel Internals – Large
- Main Output Transformer
- Steel Bricks™

Measures addressed in the Construction Program are in place that define OPG's contract management and oversight responsibilities. Contractors maintain a defined management system that is compliant with the current standards. OPG's oversight ensures that the required quality, health, safety, and security of the public and workers, and protection of the environment are maintained. OPG maintains records of contractor oversight activities and reports contractor performance that affects or has the potential to affect the quality of construction and future operational safety to regulators.

The Maintenance Program describes the processes for planning, monitoring, scheduling, and executing maintenance work activities performed during the construction and commissioning phases. During construction and commissioning (prior to fuel load) the maintenance, surveillance and in-service testing of components and systems will be managed by the design authority with the oversight and concurrence of OPG. This section describes the activities during construction and commissioning to ensure that maintenance, surveillance, inspection, and testing can be carried out effectively when the facility enters the operating phase.

The Maintenance Program (operations phase) is discussed in Chapter 13, Section 13.3.



NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

A chemistry control program, specifications, or procedures, established by the constructor, is expected to be in place to cover the requirements/aspects for the construction and commissioning phases of the project. The Chemistry Control Program establishes processes used to control contaminants to maintain system integrity during construction and commissioning.

Included in the constructor Chemistry Control Program are requirements for:

- Chemistry controls under layup conditions
- Chemistry control requirements during fabrication (provided in procurement/material specifications)
- Procedures for chemistry parameter selection, monitoring, and trending during layup and prior to startup

And the following Chemical Control Program requirements:

- Administrative controls for controlling products in the workplace
- Training
- Procedures for the storage and handling of chemicals
- Approval, procurement, and receipt of chemicals
- Listing of chemicals approved for site use, those that are precluded from site use and specifics on usage

The constructor is expected to use a quality management system planned and developed in compliance with contract requirements consistent with CSA N286 (Reference 14.4-1).

The OPG organization has oversight responsibility for Chemistry and Chemical Control Programs during the construction and commissioning phases.

The Chemistry and Chemical Control Programs, as applicable to post commissioning are described in Chapter 13, Subsection 13.3.2.

#### **14.2.4.1 Construction Plan**

The BWRX-300 design philosophy of “simplicity” and “designed for constructability” enables it to be constructed and commissioned in a short period of time.

Construction Plan details are expected to be provided in the OPG/constructor Construction Management Plan to outline compliance with CNSC REGDOC-2.3.1 (Reference 14.4-2).

#### **14.2.4.2 Planning, Scheduling, and Construction Sequence**

The project schedule utilizes the Critical Path Method technique for scheduling. The project schedule is developed in accordance with OPG and GEH standards and the project contract.

The construction Level 1 – Project Management Plan schedule provides a full set of activities and milestones with a goal of providing up to 300 MWe (nominal) to the grid by the end of 2028.

In addition, to the Level 1 plan a Level 2 – Construction, Mechanical Completion, and System Turnover Plan and a Level 3 – Engineering and Procurement Plan are developed. The project contract is expected to outline the process for modifying the Level 2 plan schedule requiring mutual agreement to modify the dates.

The Level 1 schedule is updated upon the notice to proceed and maintained monthly throughout the execution of the project.

The Level 3 schedule is routinely updated and reviewed for engineering and procurement, and also for construction and startup phase work. The progress reporting plan is aligned with OPG schedule management practices.

Overall schedule performance reporting is performed monthly with some specified weekly reporting requirements and daily updates to the schedule during construction for visibility of critical path activities.

The sequence of construction is described in the approved Construction Plan and associated schedules.

#### **14.2.4.3 Procurement and Receipt of Materials**

Processes and procedures are established to ensure equipment supplied is manufactured under a QA program that includes inspection for proper fabrication, cleanliness, calibration, and verification of operability. These processes and procedures are applicable to the construction and commissioning phases and as applicable, are continued in the operating phase within the QA program.

The Project Procurement Plan and Procurement Manual establish the purchasing process for the construction phase.

Procurement packages consisting of commercial and technical sections with any required drawing manifests are assembled into a procurement package in accordance with approved procurement processes that is reviewed by the team prior to release. OPG is provided a copy of the bid issue technical specifications. The product technical specifications establish the specific product requirements. Instructions for packaging are provided within the purchase order that describe the protection necessary during handling and transportation. If packaging and handling requirements are specified by contract, compliance is verified upon supplier notification that the material is ready for shipment. Material and equipment are categorized using a ranking system according to the consequence of failure and the probability of failure (relative risk) to determine which materials or equipment is subject to baseline assessment. The categorization uses a Level 1, 2, 3, or 4 ranking system. Criticality assessments consider the following as a minimum:

1. Consequence of equipment failure (including safety considerations, operational significance, economic significance, etc.)
2. Probable occurrence of failure
3. Complexity of the design, manufacturing process, and installation (including considerations for first of a kind designs)

Criticality ratings are established as Levels 1 and 2 (High), Level 3 (Medium), and Level 4 (low). The criticality rating is based on evaluation in five major categories: 1) plant operation, 2) financial consequences, 3) safety factors, 4) schedule (component failure), and 5) design complexity/proven technology. Risk assessment is performed on newly created specifications using a risk-based criticality assessment.

The criticality rating of the component or material sets the level of inspection requirement during manufacture.

Level 4 – Inspections may not be required or are limited to a final inspection and review of the documentation as needed to satisfy purchase specification requirements (final visual and dimensional inspection, review of testing results, positive material identification, and review of vendor data and documentation).

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Level 3 – In addition to Level 4 inspection considerations, an order review with the vendor may be done at the initial visit. Limited event-driven, in-process inspection points may be established to satisfy purchase specification requirements (e.g., initial fit-up inspection, verification of welder qualifications, review of non-destructive examination requirements, witness testing, final visual and dimensional inspections, and review of vendor data).

Level 2 – In addition to Level 3 and Level 4 considerations, the inspector's initial visit includes a detailed order review with the vendor and may include a pre-fabrication meeting. Designated witness and hold points are established in the vendor's Quality Plan/Supplier Inspection Test Plan. Level 2 inspections require multiple, event-driven, in-process inspections. The inspection process may include progressive in-process inspections up to and including dimensional checks, visual examinations, witnessing of functional or performance testing, final acceptance, and pre-shipment inspection.

Level 1 – In addition to Level 2, 3, and 4 considerations, an initial pre-fabrication meeting with formal notification may be required. Designated witness and hold points at key manufacturing points are established in the vendor's Quality Plan/Supplier Inspection Test Plan. Intensive, event-driven visits up to and including establishment of a resident inspector may be required. The inspection process may include comprehensive, progressive in-process inspections, including dimensional checks, visual examinations, witnessing of functional or performance testing, final acceptance, and pre-shipment inspection. Sub-supplied components may require inspection prior to incorporation into the final product; inspections may be required at sub-supplier fabrication locations.

Surveillance activities are planned, monitored, performed, and reported in accordance with purchase order requirements or as deemed necessary by project management.

Procurement category inspection test plans are created and maintained by qualified personnel as needed in accordance with specification baseline supplier history, equipment use, consequence of potential failure, and supplier inspection test plans.

Components receive an initial on-site "receipt inspection" to ensure the components are as ordered, have not been damaged, and that the components are not fraudulent, counterfeit or suspect. Before acceptance for use, the component is further inspected to confirm:

- Correct configuration
- Correct identification and markings
- Manufacturing and assembly documentation (including deviations) is provided
- Associated Inspection records/certificates are traceable
- Source verification release notes for components and documentation are available, if required
- Protective covers and seals are intact
- Coatings and preservatives are not damaged
- No physical damage
- Cleanliness codes and design requirements are met
- Desiccants and inert gas blankets, where relevant, are not compromised
- Receipt inspection and detected manufacturing non-conformances to be corrected on-site are recorded.

Controls are established to prevent the inadvertent installation or use of safety class components.

Processes and procedures are established to ensure that equipment supplied is manufactured under a QA program that includes inspection for proper fabrication, cleanliness, calibration, and verification of operability.

#### **14.2.4.4 Protection of Structures, Systems, and Components**

Procurement procedures document the requirements associated with maintenance and conformity of materials and equipment during on-site fabrication and the construction cycle until turnover to OPG. Preservation requirements are specified by the parties responsible for conducting the preservation and are also defined in supplier submittals.

Measures are established to protect safety class SSC from construction activities that include:

1. Preventive and corrective maintenance requirements, as required by the design, until operational programs are initiated
2. Fabrication/manufacturing, construction, and installation process requirements that do not adversely affect ageing performance
3. Periodic monitoring of environmental conditions requirements that may apply
4. Housekeeping, cleanliness, and foreign material exclusion requirements to protect sensitive mechanical, electrical and control equipment from internal and external contamination

The following equipment protections controls are established:

1. Environmental condition limits for temperature, pressure, humidity, dust, dirt, airborne salt, wind, and electromagnetic conditions as determined by the component or system design criteria
2. Foreign material exclusion measures that prevent the introduction of outside materials, debris, tools, and components where the pose a health and safety hazard or environment impact
3. Protection requirements for installed components from personnel traffic, weather, adjacent construction activity or temporary structures
4. Implementation of system specific requirements and cleaning methods
5. Compatibility requirements for cleaning methods and materials with the components being cleaned to include cleanliness requirements before installation
6. Chemistry requirements for layup, cleaning, flushing, and conditioning of piping systems and components
7. Requirements for the removal of waste material and consumables generated during construction after completion of work

Any temporary use of safety class SSC requires authorization, and the use must perform within the component's designed safety conditions.

#### **14.2.4.5 Storage**

Components are stored in accordance with design and manufacturer guidance with the following considerations:

- Cleanliness and housekeeping practices
- Fire protection requirements
- Protective coatings, preservatives, cover and sleeves
- Physical damage prevention
- Environmental control
- Preventive maintenance requirements and in-storage maintenance
- Security against theft, vandalism, and unauthorized use or alteration
- Shelf life
- Component identification

A storage log is maintained by the engineering service/constructor that documents the proper storage and maintenance of equipment and materials to ensure compliance with contract and vendor recommendation requirements. The storage logs are periodically inspected to verify the records are being properly kept. In addition, the equipment and materials are periodically inspected to ensure that preventive maintenance is being performed. Safety Data Sheets are forwarded to the site Safety Manager for retention.

#### **14.2.4.6 On-site Manufacturing and Testing**

On-site manufacturing is located where it will not affect construction activity or safety class SSC. On-site manufacturing includes (as applicable):

- Concrete strength testing
- Rebar assembly
- Pipe spool fabrication
- Modular assemblies including Steel Bricks™
- Other on-site activities that facilitate construction

Rules and procedures are established for on-site testing to ensure that industry codes and standards are met. On-site testing to include (as applicable):

- Concrete mix
- Concrete strength
- Welded joint quality
- Process instrumentation

### **14.2.5 Work Turnover**

#### **14.2.5.1 Turnover During Construction**

Construction testing commences with the completion, or partial completion, of system/component installation and terminates at pre-operational testing for the respective system/component. The specifics of construction testing are defined in the installation specifications or in the documentation provided by the major equipment suppliers. The purpose is to demonstrate that components and systems are correctly installed, calibrated to ensure accuracy, and operational, that is ready for the application of energy. These tests may include, but are not limited to, flushing and cleaning, hydrostatic testing, initial calibration of instrumentation, checks of electrical wiring and equipment, valve testing, and initial energization and operation of equipment and systems. Completion of construction testing assures systems are ready for pre-operational testing. When construction testing is completed, also referred to as "check and test," equipment and systems are turned over to the Test Group for pre-operational testing. Pre-operational testing carries out a series of tests to demonstrate that the equipment and plant operating capacity, performance, and reliability are within the prescribed limits.

Process and procedures are established to control and coordinate the turnover of completed work and associated configuration information during construction. Transfer requirements and responsibilities are documented including with access control for safety class SSC and the associated work area established and implemented for the transfer. The SSC and work areas, as well the facility configuration information, are confirmed and verified. Deficiencies are addressed prior to turnover with work or corrective actions required to be performed by the previous owner, authorized by the current owner.

#### **14.2.5.2 Turnover to Operations/Commissioning**

Processes and procedures are established that control and coordinate the turnover of work, structures, equipment, and systems when completed and the associated configuration documentation. The transfer requirements and responsibilities are documented. Procedures are in place for the transfer and ownership of SSC and the reactor facility from the design authority, construction organization and non-OPG commissioning staff to the OPG startup and operating organization.

As construction test activities are completed, equipment and systems are turned over to the Test Group for Pre-operational and Startup Test Programs.

Turnover of SSC from one organization to another is conducted as follows:

1. One organization is designated as the lead organization and ensures that all responsibilities and limits of authority are clearly established, documented, and communicated.
2. Boundaries between SSC are clearly identified in the field and on documents.
3. System status is defined.
4. Prior to acceptance, workers perform walk downs to the extent necessary on the SSC that are being turned over to ensure that they are in the state defined in the turnover documentation but not necessarily ready for operation.
5. Incomplete items, exceptions, and completion schedules are identified and listed for resolution prior to final acceptance.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

The processes and procedures controlling turnover include the following aspects:

1. Review of the facility configuration information relating to SSC, and areas by the party turning over the work and the party receiving it for completeness and accuracy
2. Performance of tests to ensure the SSC have been manufactured, constructed, and installed to confirm to design specifications
3. Identification and assessment of any remaining non-conformances or incomplete components, to ensure there is no safety implication during commissioning activities
4. Development of inaugural or baseline inspection data for systems or components for comparative purposes for in-service inspection
5. Agreement upon, planning, and scheduling of any outstanding work
6. Identification of termination points of the boundaries of turned over SSC (or parts thereof) in turnover documentation with associated required configuration
7. Inspection of turned over components and associated records and documents
8. Verification of compatibility with information and communication technology systems when turning over electronic documents and records
9. Documentation of the turnover of responsibilities including transition of maintenance
10. Establishment and turnover of approved as-built plans together with adequate and precise plant configuration details
11. Marking and tagging of all SSC turned over

Equipment belonging to Construction is governed by Construction's managed system.

Turnover from construction to commissioning is accomplished via Construction Completion Declarations. The acceptance of a Construction Completion Declaration is a prerequisite for allowing commissioning by Operations, where required, to proceed. Commissioning is the process during which an SSC is tested and verified per design requirements. Operations is the plant configuration after successful commissioning and acceptance/turnover of the SSC and is where the SSC is used/operated in accordance with operating requirements.

Prior to first system turnover, Operations establishes the Controlling Authority. Computer software for equipment status monitoring is used by the Controlling Authority to track ownership of systems turned over to Operations and all terminal points with Construction along with the status of each component. Field demarcation is also used to identify equipment turned over to Operations. Following turnover to Operations, equipment may only be operated with Operations approval and must be operated by qualified staff.

During the design phase, formal design authority rests with the organization (GEH) that has overall responsibility for the design. Prior to fuel load, this authority is transferred to OPG.

Processes and procedures are established that control and coordinate the turnover of work, structures, equipment, and systems when completed and the associated configuration documentation. The transfer requirements and responsibilities are documented. Procedures are in place for the transfer and ownership of SSC and the reactor facility from the design authority, construction organization and non-OPG commissioning staff to the OPG operating organization.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Prior to fuel-in-core testing, all SSC important to safety are under the control of OPG Operations. OPG maintains responsibility for safety and security at all times during the transfer. The transfer of SSC is documented, and all commissioning records are turned over to the OPG records management program to be retained for the lifetime of the facility.

Any equipment turned over to Operations may only be operated with specific Operations approval and must be operated and maintained by qualified staff using approved procedures. Any equipment belonging to Construction will be governed by Construction's managed system.

The above transition principle applies to the entire managed system including design authority. For systems turned over to Operations, the OPG management system will apply.



### **14.3 Commissioning**

#### **14.3.1 Organization**

The commissioning organization plans, organizes, coordinates, and maintains the status of deliverables associated with the turnover of the new facility. The commissioning organization is a multi-disciplinary team with individuals from the various organizations including construction contractor(s), GEH, equipment vendors, licensee, and others.

##### **14.3.1.1 Roles and Accountability**

The following is a generic overview of the roles and responsibilities of key individuals associated with the Commissioning Program:

1. Director Operations and Maintenance

The licensee's Director Operations and Maintenance is responsible for the implementation of the Commissioning Program. The Director Operations and Maintenance accepts responsibility of transferred systems and ensures their safe operation and maintenance.

2. Manager Turnover and Commissioning

The licensee's Manager Turnover and Commissioning reports to the Director Operations and Maintenance. The Manager Turnover and Commissioning is responsible for creating the organization that will oversee the turnover and commissioning of the new facility and for providing direction to that organization. The Manager Turnover and Commissioning is responsible for ensuring a smooth transition from the construction organization to the operating organization.

#### **14.3.2 Commissioning Management**

OPG is responsible for the identification of the health and safety, environmental, and other requirements applicable to commissioning activities and the communication of relevant requirements to all parties. The relevant requirements are considered and implemented via the established OPG management practices and controls.

The authorities and responsibilities of individuals and groups performing commissioning activities are clearly specified and delegated. The operating organization is responsible for the quality of construction activities, providing commissioning activity completion data (comprehensive baseline data and documentation) and providing qualified operations personnel for the commissioning process.

During and following commissioning, OPG is responsible for the following aspects, directly or as part of OPG's oversight responsibilities:

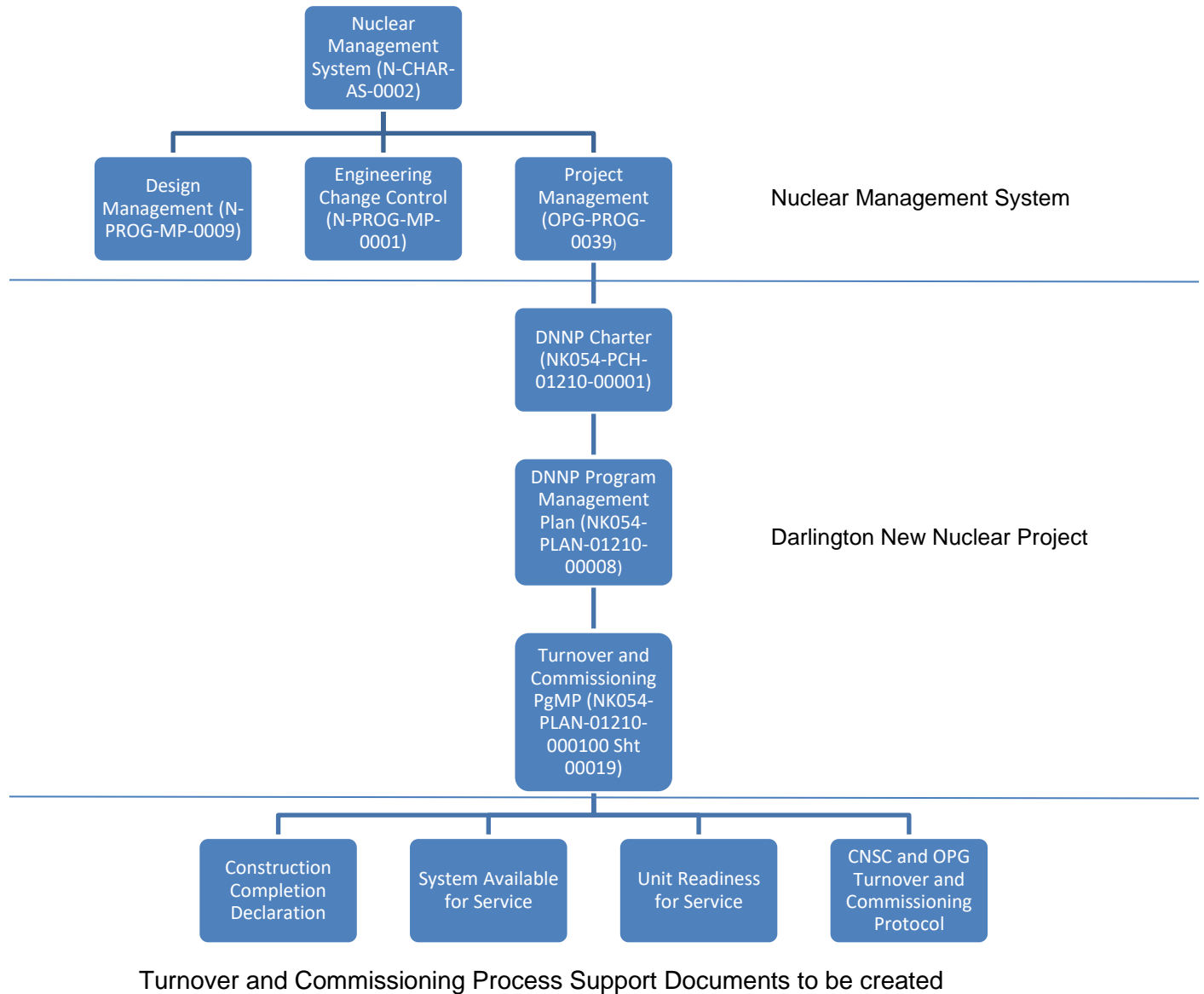
1. SSC have been constructed as per design and QA requirements are satisfied
2. SSC are tested to provide assurance that the reactor facility has been properly designed, constructed, and is ready for safe operation
3. SSC operation in accordance with the assumptions and intent of the Commissioning Program with respect to the operating limits and conditions that apply to each testing phase
4. Management, operation and maintenance of facility, systems, and components with sufficient numbers of qualified workers

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

5. Transferred systems compliance with the specified performance, design intent, and safety case
6. Documented specification of the responsibilities of other participants (designers, manufacturers, constructors, and supporting technical organizations)

The "Darlington New Nuclear Turnover and Commissioning Program Management Plan," NK054-PLAN-01210-000100 Sheet 0019 (Reference 14.4-36), describes the processes, procedures, and organization used to manage the turnover and commissioning of the facility. The program implements the applicable aspects of the OPG Nuclear Management System and is consistent with the guidance of CSA N286 (Reference 14.4-1) and CNSC REGDOC-2.3.1 (Reference 14.4-2). The framework of the plan is described in Figure 14.3.2-1.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 14.3.2-1: Turnover and Commissioning Program Management Plan Framework**

Interface arrangements and communication between OPG, commissioning organization, operations, the CNSC and other organizational groups (construction, design, contractors, etc.) performing work are established, documented, and controlled based on collaborative agreements among OPG, GEH and the constructor with construction and commissioning OPEX taken into consideration.

Processes and procedures are established for the reporting and analysis of abnormal events, human errors, and near misses with input from OPEX incorporated as part of the procedure development. Experiences are fed back into commissioning and operating personnel training programs and considerations addressed with respect to needed changes to the design and related documents.

Turnover and Commissioning Program performance indicators are established and monitored in accordance with the Darlington New Nuclear Project Turnover and Commissioning Program Management Plan.

### **14.3.3 Commissioning Program**

The Commissioning Program covers the range of activities from completion of installation work to reactor power ascension to 100%. The Commissioning Program covers the integrated plant and all SSC consistent with CNSC REGDOC-2.3.1 (Reference 14.4-2). SSC are tested to provide assurance that the facility has been properly designed and constructed and is ready for safe operation. Commissioning is accomplished in accordance with the OPG Nuclear Operations Program and NK054-PLAN-01210-000100 Sheet 0019 (Reference 14.4-36).

The Commissioning Program:

1. Defines clear responsibilities for commissioning activities and oversight
2. Is structured such that objectives and methods of testing are understood to allow management control and coordination
3. Outlines testing performed to ensure that SSC are built as designed and meets the safety analysis requirements
4. Verifies safety analysis assumptions, satisfaction of design requirements and the presence of adequate safety and operating margins
5. Ensures tests are only conducted if the reactor facility remains with the range of assumptions made in the safety analysis and the licensing basis remains valid
6. Includes the provision of temporary equipment and utilities that may be controlled by temporary modifications
7. Identifies security systems to be commissioned before nuclear fuel or material is brought on-site
8. Documents test results and identifies any impact on or changes made to the facility design
9. Validates operating and emergency procedures
10. Ensures integrated system validation of control rooms and control areas
11. Ensures a schedule including milestones and regulatory hold points, and test results to be submitted for review are identified and communicated to the CNSC

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

The process for commissioning SSC is established and controlled to confirm that the design and safety analysis requirements are met prior to placing them in service. During commissioning of SSC, they are operated and maintained within the safe operating envelope and in accordance with documentation consistent with the design. A system of permits, tags, or equivalent controls are in place to support safe operation to include the marking of the boundary of commissioning, construction, and operational activities.

SSC are commissioned in accordance with written specifications and work procedures that clearly identify the test objectives, required performance data, acceptance criteria and prerequisites for commissioning. Interface agreements between GEH and OPG define the required types of commissioning documentation and the accountability as to which party prepares, reviews, and approves the different types of documents. The interface agreements will be managed as per OPG's Design Management program, N-PROG-MP-0009, "Design Management," (Reference 14.4-37).

The management of the regulatory interface between the CNSC and OPG is described in the Licensing Program Management Plan, NK054-PLAN-01210-00100 Sheet 00008 (Reference 14.4-32).

Commissioning documentation is verified for design conformity and commissioning results are reviewed and confirmed to be acceptable. Commissioning results are incorporated into operating documentation as appropriate. Commissioning work procedures describe the specific commissioning activities and contain:

- Precautions relative to the activities to be performed
- Back-out provisions to place the nuclear power plant in a safe condition for all anticipated risks to plant and workers
- Identification of characteristics to be inspected or tested and the conditions to be controlled
- Sequential actions to be followed, including coordinating construction, commissioning, operations
- Verification activities, and hold points to be used
- Acceptance criteria to be used
- Special equipment requirements to be used
- Data to be collected

Operating organization functions are demonstrated as part of the Commissioning Program. The demonstrated functions include:

- Management
- Personnel Training
- Radiation Protection Program
- Waste Management
- Records Management
- Fire Safety
- Physical Protection
- Emergency Plan

The administrative controls of the Commissioning Program are described in NK054-PLAN-01210-000100 Sheet 0019 (Reference 14.4-36).

In addition to GEH experience related to the commissioning and startup of new nuclear power plants, the plant OPEX from organizations such as the World Association of Nuclear Operators, Institute of Nuclear Power Operations, and Electric Power Research Institute is reviewed and applied to the development of the initial Commissioning Program of the BWRX-300.

Startup testing begins with fuel load and continues until the commercial operation milestone is met.

#### **14.3.4 Commissioning Tests**

Significant testing for commissioning is performed at the completion of construction activities to ensure that each SSC works individually and as integrated within its respective system(s). Prior to fuel load, each component and system of the BWRX-300 is tested to confirm that it will perform properly during operations. Once individual systems are tested, integrated system testing is performed with testing results evaluated for acceptability.

The activities performed in commissioning may be divided into the following categories:

1. Construction and installation testing to ensure that SSC have been manufactured, constructed, and installed according to design specifications
2. Pre-operational testing of systems prior to fuel load to confirm the operability, availability, and performance of SSC that ensure safety with fuel in the core
3. Startup testing, including initial fuel loading, subcriticality tests, initial criticality tests, low power tests, and power ascension tests to confirm reactor behavior

Testing is the core activity of the Commissioning Program and is sufficiently comprehensive to demonstrate that the facility can operate in the modes for which it has been designed. Tests necessary to demonstrate operability, safety and safety-related functions are fully performed. Where tests cannot be fully performed, documentation and the result of any alternative testing performed, is provided to demonstrate how safety and design intent has been achieved.

Brief descriptions are expected to be developed for all the commissioning tests conducted during the initial Commissioning Program, with emphasis on safety systems and safety features that are relied on for the following:

1. Safe shutdown and cool down of the plant in operational states and accident conditions
2. Conformance with Operational Limits and Conditions that will be established by the technical specifications
3. Prevention or mitigation of the consequences of anticipated operational occurrences and accident conditions

The summary of commissioning tests is expected to be included with the Licence to Operate application submission with details of all specific testing, finalized and in place prior to receipt of the Licence to Operate.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

Testing is performed under realistic operating conditions, as practicable, and confirms any analytical validations. Proposed operating and maintenance procedures are validated to the extent practicable with participation of suitably trained and qualified operations and technical staff personnel. Operating and test procedures are verified for technical accuracy and validated to ensure usability with the installed equipment and control systems. Verification and validation is performed to the extent possible prior to fuel handling operations on-site and continued during the duration of the commissioning phase. Verification and validation is also used for overall operation procedures.

Processes are also in place for performing trial use of emergency procedures during commissioning testing.

The testing is sufficiently comprehensive to provide the reference data that characterize the SSC and provides information retained to ensure plant safety and evaluation in subsequent safety reviews.

Acceptance criteria for commissioning tests are defined by test procedure and the technical basis for the criteria is documented prior to conducting the tests. When analytical tools are used for testing, the testing and criteria are consistent with CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 14.4-15).

Reviewed and approved arrangements for work control, modification control, and configuration control are established that meet the conditions of the commissioning tests. Commissioning testing is performed in accordance with procedures that have been reviewed, verified, and approved by OPG. The design authority provides the administrative controls that are used to develop, review, and approve individual test procedures, coordination with organizations involved in the test program, participation of facility operational and technical staff, and the review, evaluation, and approval of test results. Acceptance criteria for commissioning activities necessary to provide reasonable assurance that the as-built facility will conform to the approved plant design and applicable regulations are provided by the design authority.

Test results are reviewed by the commissioning organization and all deviations are resolved and operating restraints are identified and documented. Formal reports are prepared and approved by the commissioning and design organizations. Modifications to test procedures must be authorized by means of an approved process by the design authority that controls the change to documentation.

#### **14.3.5 Test Phases**

Commissioning and associated testing occurs in four basic phases with hold points to ensure the prerequisites are complete and required approvals obtained prior to transitioning from one phase to another. Hold points are controlled by the Director Operations and Maintenance, identified in the schedule, and require regulatory approval.

Criteria for the release of a hold point is addressed in the readiness for operation review. Written confirmation for hold points is provided to the CNSC that identifies the following:

1. Completion of project commitments tied to the hold point
2. Confirmation that all required system functions for safe operation beyond the hold point are available
3. Other information as appropriate

Testing is performed in a logical progressive sequence consistent with the guidance provided in CNSC REGDOC-2.3.1 (Reference 14.4-2). Review of test results for each stage is completed before commissioning continues to the next stage.

Specific testing is addressed in commissioning testing procedures. The Commissioning Program establishes the tests necessary to demonstrate the as-built, as installed plant satisfies the approved design, meets the requirements of the safety analysis report, and the plant can be operated in accordance with the operational limits and conditions.

Testing is performed in four basic phases:

- Prior to fuel load
- Prior to leaving reactor guaranteed shutdown state
- Approach to critical and low-power testing
- High power testing

#### **14.3.6 Structures, Systems, and Components and Facility Transfer**

Turnover and commissioning follow the OPG Corporate, and Nuclear Governance as described in:

- N-CHAR-AS-0002, Nuclear Management System (Reference 14.4-38)
- OPG-PROG-0039, Project Management (Reference 14.4-31)
- N-PROG-MP-0001, Engineering Change Control (Reference 14.4-39)
- N-PROG-MP-0009, Design Management (Reference 14.4-37)

Procedures are expected to be established for the transfer and ownership of SSC and the reactor facility to the OPG operating organization with detailed process steps including a description of the responsibilities and authorities of the parties involved. OPG is responsible for safety and security at all times during the transfer. The transfer of SSC is documented with all commissioning records turned over to the OPG operating organization records management program for life of the reactor facility retention.

Upon completion of the Commissioning Program, records relating to commissioning procedures and test data are transferred to the licensee's approved information management system with the appropriate retention schedule applied. Documentation is transferred in system packages over a reasonable time dependent on how responsibilities for testing are assigned and to allow a comprehensive review of every package.



#### **14.4 References**

- 14.4-1 CSA N286, "Management System Requirements for Nuclear Facilities," CSA Group.
- 14.4-2 CNSC Regulatory Document REGDOC-2.3.1, "Conduct of Licensed Activities: Construction and Commissioning Programs."
- 14.4-3 CNSC Regulatory Document REGDOC-2.12.1, "High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices."
- 14.4-4 CNSC Regulatory Document REGDOC-2.12.3, "Security of Nuclear Substances: Sealed Sources and Category I, II, and III Nuclear Material," September 2020.
- 14.4-5 CNSC Regulatory Document REGDOC-2.12.2, "Security: Site Access Security Clearance."
- 14.4-6 CSA N290.7, "Cyber Security for Nuclear Facilities," CSA Group.
- 14.4-7 CNSC Regulatory Document REGDOC-2.2.4, "Fitness for Duty, Volume II; Managing Alcohol and Drug Use," January 2021.
- 14.4-8 Government of Canada SOR/2000-202, "General Nuclear Safety and Control Regulations."
- 14.4-9 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."
- 14.4-10 CNSC Regulatory Document REGDOC-2.2.2, "Personnel Training."
- 14.4-11 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 14.4-12 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 14.4-13 CSA N286.10, "Configuration Management for High Energy Facilities," CSA Group.
- 14.4-14 CNSC Regulatory Document REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants."
- 14.4-15 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 14.4-16 OPG-PROG-0038, "Contract Management," Ontario Power Generation.
- 14.4-17 NK054-PLAN-01210-00100 Sheet 0003, "Darlington New Nuclear Project Quality Program Management Plan," Ontario Power Generation.
- 14.4-18 NK054-PLAN-61400-00001, "DNNP Site Security Plan," Ontario Power Generation.
- 14.4-19 OPG-STD-0030, "Protecting OPG's Information," Ontario Power Generation.
- 14.4-20 NK054-PLAN-01210-00100 Sheet 0007, "Darlington New Nuclear Project Training Program Management Plan," Ontario Power Generation.
- 14.4-21 NK054-PLAN-01210-00008, "Darlington New Nuclear Project - Program Management Plan," Ontario Power Generation.
- 14.4-22 N-PROG-TR-0005, "Training," Ontario Power Generation.
- 14.4-23 N-TQD-603-00001, "Nuclear Security Training and Qualification Description," Ontario Power Generation.

NEDO-33964 REVISION 0  
NON-PROPRIETARY INFORMATION

- 14.4-24 N-PROC-TR-0044, "Training Demand, Scheduling, and Cancellation Process," Ontario Power Generation.
- 14.4-25 N-PROC-TR-0041, "TIMS II Administration," Ontario Power Generation.
- 14.4-26 N-PROC-TR-0012, "Records and Documentation," Ontario Power Generation.
- 14.4-27 N-PROG-AS-0001, "Nuclear Management System Administration," Ontario Power Generation.
- 14.4-28 OPG-PROG-0001, "Information Management," Ontario Power Generation.
- 14.4-29 N-PROC-AS-0028, "Development, Review and Approval of Technical Procedures," Ontario Power Generation.
- 14.4-30 NK054-PLAN-01210-00100 Sht. 00009, "Darlington New Nuclear Project Construction Program Management Plan," Ontario Power Generation.
- 14.4-31 OPG-PROG-0039 "Project Management," Ontario Power Generation.
- 14.4-32 NK054-PLAN-01210-00100 Sheet 00008, "Darlington New Nuclear Project - Licensing Program Management Plan," Ontario Power Generation.
- 14.4-33 OPG-STD-0148, "Project Management," Ontario Power Generation.
- 14.4-34 OPG-MAN-00120-0021, "Project Construction Management," Ontario Power Generation.
- 14.4-35 N-INS-00120-10008, "Nuclear Contractor Safety Management Process," Ontario Power Generation.
- 14.4-36 NK054-PLAN-01210-000100, Sheet 0019, "Darlington New Nuclear Project Turnover and Commissioning Program Management Plan," Ontario Power Generation.
- 14.4-37 N-PROG-MP-0009, "Design Management," Ontario Power Generation.
- 14.4-38 N-CHAR-AS-0002, "Nuclear Management System," Ontario Power Generation.
- 14.4-39 N-PROG-MP-0001, "Engineering Change Control," Ontario Power Generation.
- 14.4-40 CNSC Regulatory Document REGDOC-2.12.1, "High Security Facilities, Volume I: Nuclear Response Force."



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33965

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 15  
Safety Analysis**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AAZ	Automatic Action Zone
AC	Alternating Current
ACRW	All Control Rod Withdrawal at Power
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATLM	Automatic Thermal Limit Monitor
BDBA	Beyond Design Basis Accident
BE	Best Estimate
BIS	Boron Injection System
BL-AOO	Baseline Abnormal Operational Occurrence
BL-DBA	Baseline Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CANDU	CANada Deuterium Uranium
CB	Control Building
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CIV	Containment Isolation Valve
CN-DBA	Conservative Design Basis Accident
CN-DSA	Conservative Deterministic Safety Analysis
CNSC	Canadian Nuclear Safety Commission
CPR	Critical Power Ratio
CRD	Control Rod Drive

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Motor
CSA	CSA Group
CSAU	Code Scaling, Applicability, and Uncertainty
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DF	Decontamination Factor
DL	Defense Line
D-in-D	Defence-in-Depth
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHE	External Hazard Evaluation
EOC	End of Cycle
EOP	Emergency Operating Procedure
EOR	End of Rated Cycle
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Accident
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FFHE	Functional Failure Hazard Evaluation
FHA	Fuel Handling Accident
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
FSF	Fundamental Safety Function
FV	Fussell-Vesely
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWFI	Feedwater Flow Increase
FWPT	Feedwater Pump Trip
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HEP	Human Error Probabilities
HFE	Human Factors Engineering
HOHE	Human Operation Hazard Evaluation
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal at Power – Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
IR	Inventory Reduction
LFWH	Loss of Feedwater Heating
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LPSD	Low Power Shutdown
LPZ	Low Population Zone
LRF	Large Release Frequency
LR-TT	Load Reduction – Turbine Trip
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MCCI	Molten Core Concrete Interaction
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
MCS	Minimal Cutset
MOC	Middle of Cycle
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
NBR	Nuclear Boiler Rated
NPP	Nuclear Power Plant
OBE	Operating-Basis Earthquake
ODE	Ordinary Differential Equation
OLC	Operational Limits and Conditions
OLMCPR	Operating Limit Minimum Critical Power Ratio
OCC	Out Of Core Criticality
OPEX	Operating Experience
OPG	Ontario Power Generation
PA	Postulated Accident
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCW	Plant Cooling Water System
PI-AOO	Pressure Increase- Abnormal Operational Occurrence
PIE	Postulated Initiating Event
PIRT	Phenomena Identification and Ranking Table
POS	Plant Operating State
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objectives

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Acronym</b>	<b>Explanation</b>
RAW	Risk Achievement Worth
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RI	Reactivity Increase
RI-AOO	Reactivity Increase- Abnormal Operational Occurrence
RIV	Reactor Isolation Valve
RPC	Reactor Pressure Control
RPF	Radial Peaking Factor
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SSC	Structures, Systems, and Components
SS-DBA	Safe-Shutdown Design Basis Analysis
STP	Simulated Thermal Power
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TRACG	Transient Reactor Analysis Code General Electric
TSV	Turbine Stop Valve
URD	Utility Requirements Document
USNRC	U.S. Nuclear Regulatory Commission
$\Delta$ CPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

## TABLE OF CONTENTS

<b>15.0 SAFETY ANALYSIS .....</b>	<b>15-1</b>
15.1 General Considerations of the BWRX-300 Safety Analyses.....	15-3
15.1.1 Safety Analysis Objectives, Scope, and Approach .....	15-3
15.1.2 Analysis of Hazards .....	15-4
15.1.3 Analysis of Design Basis Conditions.....	15-6
15.1.4 Analysis of Design Extension Conditions Without Core Damage .....	15-6
15.1.5 Analysis of Beyond Design Basis Accidents with Core Damage (Severe Accident) .....	15-7
15.1.6 References.....	15-7
15.2 Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios .....	15-8
15.2.1 Basis for Categorization of Postulated Initiating Events, Accident Scenarios and Fault Evaluation .....	15-9
15.2.2 Categorization of Events According to Their Frequencies .....	15-10
15.2.3 Grouping of Events According to Type .....	15-11
15.2.4 Postulated Initiating Events and Accident Scenarios .....	15-13
15.2.5 References.....	15-26
15.3 Safety Objectives and Acceptance Criteria .....	15-32
15.3.1 Deterministic Safety Analysis Acceptance Criteria .....	15-32
15.3.2 Acceptance Criteria for Probabilistic Safety Assessment .....	15-33
15.3.3 References.....	15-33
15.4 Human Actions.....	15-39
15.4.1 Human Actions in Deterministic Safety Analysis.....	15-39
15.4.2 Human Actions in Probabilistic Safety Assessment.....	15-39
15.5 Deterministic Safety Analysis.....	15-40
15.5.1 General Description of the Approach .....	15-40
15.5.2 Analysis of Normal Operation .....	15-48
15.5.3 Analysis of Anticipated Operational Occurrences .....	15-50
15.5.4 Analysis of Design Basis Accidents .....	15-61
15.5.5 Analysis of Design Extension Conditions Without Core Damage .....	15-85
15.5.6 Analysis of Design Extension Conditions with Core Damage .....	15-98
15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool .....	15-98
15.5.8 Analysis of Fuel Handling Events.....	15-99

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

15.5.9	Analysis of Radioactive Releases from a Subsystem or a Component .....	15-103
15.5.10	Analysis of Internal and External Hazards .....	15-113
15.5.11	References .....	15-114
15.6	Probabilistic Safety Assessment .....	15-116
15.6.1	General Approach to Probabilistic Safety Assessment.....	15-116
15.6.2	Results of the Level 1 Probabilistic Safety Assessment .....	15-171
15.6.3	Probabilistic Safety Assessment Insights and Applications .....	15-173
15.6.4	References.....	15-175
15.7	Results of the Deterministic Safety Analyses and Probabilistic Safety Assessment.....	15-192
15.7.1	Results of Analysis of Normal Operation .....	15-192
15.7.2	Results of Analysis of Anticipated Operational Occurrences .....	15-192
15.7.3	Results of Analysis of Design Basis Accidents .....	15-192
15.7.4	Results of Analysis of Design Extension Conditions Without Core Damage .....	15-192
15.7.5	Results of Analysis of Design Extension Conditions with Core Damage .....	15-193
15.7.6	Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool.....	15-193
15.7.7	Results of Analysis of Fuel Handling Events.....	15-193
15.7.8	Results of Analysis of Radioactive Releases from a Subsystem or a Component .....	15-193
15.7.9	Results of Analysis of LOCA Breaks Outside Containment.....	15-193
15.7.10	Results of Analysis of Internal and External Hazards .....	15-193
15.7.11	Results of Probabilistic Safety Analysis .....	15-194
Appendix 15A.	Reference Source Term for Conditions That Are Practically Eliminated .....	15-203
15A.1	Safety Objectives .....	15-203
15A.2	Acceptance Criteria.....	15-203
15A.3	Approach.....	15-203
15A.4	Practically Eliminated Events.....	15-204
Appendix 15B.	Complementary Design Features for Mitigating Design Extension Conditions .....	15-210
15B.1	Boron Injection System .....	15-210

## LIST OF TABLES

Table 15.2-1: Fault or Event Groups and Explanation of Core Reactivity Response Basis...	15-27
Table 15.2-2: Bounding Events Transient (Non-LOCA) and LOCA.....	15-28
Table 15.3-1: Anticipated Operational Occurrence Deterministic Safety Assessment Acceptance Criteria .....	15-34
Table 15.3-2: Design Basis Accident Acceptance Criteria.....	15-36
Table 15.3-3: Probabilistic Safety Goals .....	15-38
Table 15.6-1: Internal Hazard Qualitative Screening Analysis.....	15-178
Table 15.6-2: Internal Hazard Quantitative Screening Analysis .....	15-179
Table 15.6-3: External Hazards Qualitatively Screened Out of the PSA .....	15-180
Table 15.6-4: External Hazards Quantitative Assessment Summary .....	15-181
Table 15.6-5: Generic List of Boiling Water Reactor Transient Initiating Events .....	15-183
Table 15.6-6: Level 1 PSA Damage Class Definitions.....	15-185
Table 15.6-7: Internal Initiating Events At-Power.....	15-186
Table 15.6-8: Internal Initiating Events At-Power and Shutdown.....	15-188
Table 15.7.2-1: Results Summary of AOO Events .....	15-195
Table 15.7.3-1: Results Summary of the DBA and DEC Events – Non- LOCA.....	15-196
Table 15.7.3-2: Results Summary of Fuel Loading Error Event .....	15-198
Table 15.7.3-3: Results Summary of DBA and DEC Events For LOCA .....	15-198
Table 15.7.6-1: Fuel Handling Accident (DEC) Dose Consequences .....	15-199
Table 15.7.7-1: BWRX-300 DNNP Main Steam Line Break Outside Containment Accident Dose Consequences .....	15-199
Table 15.7.7-2: BWRX-300 DNNP Feedwater Line Break Outside Containment Accident Dose Consequences .....	15-200
Table 15.7.7-3: BWRX-300 DNNP Isolation Condenser System Line Break Outside Containment Accident Dose Consequences .....	15-200
Table 15.7.7-4: BWRX-300 DNNP Instrument Line Break Outside Containment Accident Dose Consequences .....	15-201
Table 15.7.9-1: Core Damage Frequency Results .....	15-201
Table 15.7.9-2: Large Release Frequency Results .....	15-202
Table 15A-1: Practically Eliminated Conditions .....	15-205
Table 15B-1: Complementary Design Features Mitigating Functions.....	15-217
Table 15B-2: Boron Injection System Interfaces.....	15-219
Table 15.5-1: Key Initial Conservative LOCA Evaluations.....	15-222
Table 15.5-2: Summary of Core-Wide Decay Ratio Results.....	15-222

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Table 15.5-3: Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses .....	15-223
Table 15.5-4: CRD Scram Time.....	15-225
Table 15.5-5: Defense Lines Inputs Used in Non-LOCA Analyses.....	15-226
Table 15.5-6: Sequence of Events for Loss of Feedwater Heating (AOO) .....	15-229
Table 15.5-7: Sequence of Events for Turbine Trip (AOO).....	15-229
Table 15.5-8: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (AOO) .....	15-229
Table 15.5-9: Sequence of Events for Loss of Condenser Vacuum (AOO).....	15-230
Table 15.5-10: Sequence of Events for Loss-of-Preferred Power (AOO).....	15-230
Table 15.5-11: Sequence of Events for Feedwater Pump Trip (AOO) .....	15-230
Table 15.5-12: Sequence of Events for Inadvertent Isolation Condenser Initiation One Train (AOO) .....	15-231
Table 15.5-13: Sequence of Events for Loss of Feedwater Heating (DBA).....	15-231
Table 15.5-14: Sequence of Events for Generator Load Rejection (DBA) .....	15-231
Table 15.5-15: Sequence of Loss-of-Preferred Power (DBA).....	15-232
Table 15.5-16: RPV Pressure Control Downscale (DBA) .....	15-232
Table 15.5-17: Closure of All MSRIVs and FW Isolation Valves (DBA).....	15-232
Table 15.5-18: Sequence of Events for Feedwater Flow Increase – All Pumps .....	15-233
Table 15.5-19: Sequence of Events for Condenser Initiation – All Trains .....	15-233
Table 15.5-20: Sequence of Events for Loss of Feedwater (DBA) .....	15-233
Table 15.5-21: Sequence of Events for RPV Pressure Control Open (DBA) .....	15-234
Table 15.5-22: Not Used .....	15-234
Table 15.5-23: Timing of Events for Main Steam Pipe Break Inside Containment, CN-DSA.....	15-234
Table 15.5-24: Timing of Events for Feedwater Pipe Break Inside Containment, CN-DSA.....	15-236
Table 15.5-25: Timing of Events for Small Steam Pipe Break Inside Containment CN-DSA.....	15-237
Table 15.5-26: Timing of Events for Small Liquid Pipe Break Inside Containment, CN-DSA.....	15-238
Table 15.5-27: Sequence of Events for Closure of One Main Steam Reactor Pressure Isolation Valve (DEC) .....	15-239
Table 15.5-28: Sequence of Events for Complex Sequence Generator Load Rejection (DEC).....	15-239
Table 15.5-29: Sequence of Events for Loss of Condenser Vacuum (DEC) .....	15-239
Table 15.5-30: Sequence of Events for Loss of Preferred Power (DEC).....	15-240

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Table 15.5-31: Sequence of Events for All Control Rod Withdrawal at Power (ACRW).....	15-240
Table 15.5-32: Sequence of Events for Inadvertent Control Rod Withdrawal at Power Single Rod (ICRW).....	15-240
Table 15.5-33: Sequence of Events for Feedwater Isolation (DEC).....	15-241
Table 15.5-34: Fuel Handling Accident Sequence of Events.....	15-241
Table 15.5-35: BWRX-300 Core Parameters .....	15-242
Table 15.5-36: BWRX-300 Core Inventory 24 Hours After Shutdown .....	15-243
Table 15.5-37: Non-LOCA Fraction of Fission Product Inventory in Gap .....	15-244
Table 15.5-38: BWRX-300 FHA Activity Released from Fuel.....	15-244
Table 15.5-39: BWRX-300 Fuel Handling Accident (FHA) Activity Released from the Reactor Cavity Pool.....	15-245
Table 15.5-40: Not Used.....	15-245
Table 15.5-41: FWLB Line Break Accident Airborne Release Source Term .....	15-246
Table 15.5-42: ICS Line Break Accident Airborne Release Source Term .....	15-248
Table 15.5-43A: Instrument Line Break (ILB) Accident Airborne Equilibrium Release Source Term.....	15-250
Table 15.5-43B: Instrument Line Break (ILB) Accident Airborne Iodine Spike Release Source Term.....	15-253
Table 15.5-44: Main Steam Line Break Accident Airborne Release Source Term .....	15-256
Table 15.5-45: Description of Criticality Accident Parameters .....	15-258
Table 15.5-46: Out of Core Criticality Scenario Results .....	15-258
Table 15.5-47: Protective Actions and Generic Criteria for OOCOC .....	15-258
Table 15.5-48: Conservatisms Used in the Non-LOCA DSA.....	15-259
Table 15.5-49: DL3 Functions Credited in Conservative LOCA Analyses .....	15-260
Table 15.5-50: DL2 and DL4a Functions Credited in DEC LOCA Analyses.....	15-261

## LIST OF FIGURES

Figure 15.6-1: Principal Steps in Probabilistic Safety Assessment.....	15-190
Figure 15.6-2: Overall Process for Selection, Screening and Analysis of Hazards .....	15-191
Figure 15B-1: BWRX-300 Sodium Pentaborate Injection System Interfaces .....	15-220
Figure 15B-2: Boron Injection System Simplified Flow Diagram .....	15-221
Figure 15.5-1: MOC Transient Results for Core-Wide Stability Analysis .....	15-261
Figure 15.5-2: MOC Transient Results for LFWH with SCCRI Core-Wide Stability Analysis .....	15-261
Figure 15.5-3: Regional Mode Stability Response at MOC for FW Temperature of 241.9°C.....	15-262
Figure 15.5-4: Not Used.....	15-262
Figure 15.5-5: Loss of Feedwater Heating (AOO) .....	15-263
Figure 15.5-6: Loss of Feedwater Heating (AOO) .....	15-264
Figure 15.5-7: Loss of Feedwater Heating (AOO) .....	15-265
Figure 15.5-8: Loss of Feedwater Heating (AOO) .....	15-266
Figure 15.5-9: Loss of Feedwater Heating (AOO) .....	15-267
Figure 15.5-10: Loss of Feedwater Heating (AOO) .....	15-268
Figure 15.5-11: Turbine Trip (AOO).....	15-269
Figure 15.5-12: Turbine Trip (AOO).....	15-270
Figure 15.5-13: Turbine Trip (AOO).....	15-271
Figure 15.5-14: Turbine Trip (AOO).....	15-272
Figure 15.5-15: Turbine Trip (AOO).....	15-273
Figure 15.5-16: Turbine Trip (AOO).....	15-274
Figure 15.5-17: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-275
Figure 15.5-18: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-276
Figure 15.5-19: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-277
Figure 15.5-20: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-278
Figure 15.5-21: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-279
Figure 15.5-22: Closure of One Main Steam Reactor Isolation Valve (AOO).....	15-280
Figure 15.5-23: Loss of Condenser Vacuum (AOO).....	15-281
Figure 15.5-23a: Loss of Condenser Vacuum (AOO).....	15-282
Figure 15.5-24: Loss of Condenser Vacuum (AOO).....	15-283
Figure 15.5-25: Loss of Condenser Vacuum (AOO).....	15-284
Figure 15.5-26: Loss of Condenser Vacuum (AOO).....	15-285
Figure 15.5-27: Loss of Condenser Vacuum (AOO).....	15-286



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-28: Loss of Preferred Power (AOO) .....	15-287
Figure 15.5-29: Loss of Preferred Power (AOO) .....	15-288
Figure 15.5-30: Loss of Preferred Power (AOO) .....	15-289
Figure 15.5-31: Loss of Preferred Power (AOO) .....	15-290
Figure 15.5-32: Loss of Preferred Power (AOO) .....	15-291
Figure 15.5-33: Loss of Preferred Power (AOO) .....	15-292
Figure 15.5-34: Feedwater Pump Trip – One Pump (AOO) .....	15-293
Figure 15.5-35: Feedwater Pump Trip – One Pump (AOO) .....	15-294
Figure 15.5-36: Feedwater Pump Trip – One Pump (AOO) .....	15-295
Figure 15.5-37: Feedwater Pump Trip – One Pump (AOO) .....	15-296
Figure 15.5-38: Feedwater Pump Trip – One Pump (AOO) .....	15-297
Figure 15.5-39: Feedwater Pump Trip – One Pump (AOO) .....	15-298
Figure 15.5-40: Inadvertent Isolation Condenser Initiation – One Train (AOO).....	15-299
Figure 15.5-41: Inadvertent Isolation Condenser Initiation – One Train (AOO).....	15-300
Figure 15.5-42: Inadvertent Isolation Condenser Initiation – One Train (AOO).....	15-301
Figure 15.5-43: Inadvertent Isolation Condenser Initiation - One Train (AOO).....	15-302
Figure 15.5-44: Inadvertent Isolation Condenser Initiation - One Train (AOO).....	15-303
Figure 15.5-45: Inadvertent Isolation Condenser Initiation - One Train (AOO).....	15-304
Figure 15.5-46: Loss of Feedwater Heating (DBA).....	15-305
Figure 15.5-47: Loss of Feedwater Heating (DBA).....	15-306
Figure 15.5-48: Loss of Feedwater Heating (DBA).....	15-307
Figure 15.5-49: Loss of Feedwater Heating (DBA).....	15-308
Figure 15.5-50: Loss of Feedwater Heating (DBA).....	15-309
Figure 15.5-51: Loss of Feedwater Heating (DBA).....	15-310
Figure 15.5-52: Generator Load Rejection (DBA).....	15-311
Figure 15.5-53: Generator Load Rejection (DBA).....	15-312
Figure 15.5-54: Generator Load Rejection (DBA).....	15-313
Figure 15.5-55: Generator Load Rejection (DBA).....	15-314
Figure 15.5-56: Generator Load Rejection (DBA).....	15-315
Figure 15.5-57: Generator Load Rejection (DBA).....	15-316
Figure 15.5-58: Loss of Preferred Power (DBA) .....	15-317
Figure 15.5-59: Loss of Preferred Power (DBA) .....	15-318
Figure 15.5-60: Loss of Preferred Power (DBA) .....	15-319
Figure 15.5-61: Loss of Preferred Power (DBA) .....	15-320

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-62: Loss of Preferred Power (DBA) .....	15-321
Figure 15.5-63: Loss of Preferred Power (DBA) .....	15-322
Figure 15.5-64: RPV Pressure Control Downscale (DBA) .....	15-323
Figure 15.5-65: RPV Pressure Control Downscale (DBA) .....	15-324
Figure 15.5-66: RPV Pressure Control Downscale (DBA) .....	15-325
Figure 15.5-67: RPV Pressure Control Downscale (DBA) .....	15-326
Figure 15.5-68a: RPV Pressure Control Downscale (DBA) .....	15-327
Figure 15.5-68b: RPV Pressure Control Downscale (DBA) .....	15-328
Figure 15.5-69: RPV Pressure Control Downscale (DBA) .....	15-329
Figure 15.5-70: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-330
Figure 15.5-71: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-331
Figure 15.5-72: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-332
Figure 15.5-73: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-333
Figure 15.5-74: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-334
Figure 15.5-75: Closure of All MSRIVs and FW Isolation Valves (DBA) .....	15-335
Figure 15.5-76: Feedwater Flow Increase (DBA) .....	15-336
Figure 15.5-77: Feedwater Flow Increase (DBA) .....	15-337
Figure 15.5-78: Feedwater Flow Increase (DBA) .....	15-338
Figure 15.5-79: Feedwater Flow Increase (DBA) .....	15-339
Figure 15.5-80: Feedwater Flow Increase (DBA) .....	15-340
Figure 15.5-81: Feedwater Flow Increase (DBA) .....	15-341
Figure 15.5-82: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-342
Figure 15.5-83: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-343
Figure 15.5-84: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-344
Figure 15.5-85: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-345
Figure 15.5-86: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-346
Figure 15.5-87: Inadvertent Isolation Condenser Initiation - All Trains (DBA) .....	15-347
Figure 15.5-88: Loss of Feedwater Flow (DBA) .....	15-348
Figure 15.5-89: Loss of Feedwater Flow (DBA) .....	15-349
Figure 15.5-90: Loss of Feedwater Flow (DBA) .....	15-350
Figure 15.5-91: Loss of Feedwater Flow (DBA) .....	15-351
Figure 15.5-92: Loss of Feedwater Flow (DBA) .....	15-352
Figure 15.5-93: Loss of Feedwater Flow (DBA) .....	15-353
Figure 15.5-94: RPV Pressure Control Open (DBA) .....	15-354

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-95: RPV Pressure Control Open (DBA).....	15-355
Figure 15.5-96: RPV Pressure Control Open (DBA).....	15-356
Figure 15.5-97: RPV Pressure Control Open (DBA).....	15-357
Figure 15.5-98: RPV Pressure Control Open (DBA).....	15-358
Figure 15.5-99: RPV Pressure Control Open (DBA).....	15-359
Figure 15.5-100: Reactor Power, Large Main Steam Pipe Break, Conservative Case.....	15-360
Figure 15.5-101: Reactor Pressure, Large Main Steam Pipe Break, Conservative Case ...	15-361
Figure 15.5-102: Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case .....	15-362
Figure 15.5-103: Reactor Water Level, Large Main Steam Pipe Break, Conservative Case.....	15-363
Figure 15.5-104: Containment Pressure, Large Main Steam Pipe Break, Conservative Case.	15-364
Figure 15.5-105: Containment Temperatures, Large Main Steam Pipe Break, Conservative Case .....	15-365
Figure 15.5-106: Reactor Power, Small Steam Break With LOPP, Conservative Case .....	15-366
Figure 15.5-107: Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case .....	15-367
Figure 15.5-108: Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case .....	15-368
Figure 15.5-109: Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case .....	15-369
Figure 15.5-110: Containment Pressure, Small Steam Pipe Break with LOPP 2 ICS Trains, Conservative Case .....	15-370
Figure 15.5-111: Containment Temperature, Small Steam Pipe Break, Conservative Case ....	15-371
Figure 15.5-112: Not Used.....	15-372
Figure 15.5-113: Not Used.....	15-372
Figure 15.5-114: Not Used.....	15-372
Figure 15.5-115: Reactor Power, Small Liquid Pipe Break, Conservative Case .....	15-373
Figure 15.5-116: Reactor Pressure, Small Liquid Pipe Break, Conservative Case .....	15-374
Figure 15.5-117: Reactor Water Level, Small Liquid Break, Conservative Case.....	15-375
Figure 15.5-118: Containment Pressure After RPV Depressurization, Small Liquid Pipe Break, Conservative Case .....	15-376
Figure 15.5-119: Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case .....	15-377
Figure 15.5-120: Containment Pressure, Small Liquid Pipe Break, Conservative Case .....	15-378

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-121: Containment Temperature, Small Liquid Pipe Break, Conservative Case .....	15-379
Figure 15.5-122: Not Used.....	15-380
Figure 15.5-123: Not Used.....	15-380
Figure 15.5-124: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-381
Figure 15.5-125: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-382
Figure 15.5-126: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-383
Figure 15.5-127: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-384
Figure 15.5-128: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-385
Figure 15.5-129: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC) .....	15-386
Figure 15.5-130: Complex Sequence Generator Load Rejection (DEC) .....	15-387
Figure 15.5-131: Complex Sequence Generator Load Rejection (DEC) .....	15-388
Figure 15.5-132: Complex Sequence Generator Load Rejection (DEC) .....	15-389
Figure 15.5-133: Complex Sequence Generator Load Rejection (DEC) .....	15-390
Figure 15.5-134: Complex Sequence Generator Load Rejection (DEC) .....	15-391
Figure 15.5-135: Complex Sequence Generator Load Rejection (DEC) .....	15-392
Figure 15.5-136: Loss of Condenser Vacuum (DEC) .....	15-393
Figure 15.5-137: Loss of Condenser Vacuum (DEC) .....	15-394
Figure 15.5-138: Loss of Condenser Vacuum (DEC) .....	15-395
Figure 15.5-139: Loss of Condenser Vacuum (DEC) .....	15-396
Figure 15.5-140: Loss of Condenser Vacuum (DEC) .....	15-397
Figure 15.5-141: Loss of Condenser Vacuum (DEC) .....	15-398
Figure 15.5-142: Loss of Preferred Power (DEC).....	15-399
Figure 15.5-143: Loss of Preferred Power (DEC).....	15-400
Figure 15.5-144: Loss of Preferred Power (DEC).....	15-401
Figure 15.5-145: Loss of Preferred Power (DEC).....	15-402
Figure 15.5-146: Loss of Preferred Power (DEC).....	15-403
Figure 15.5-147: Loss of Preferred Power (DEC).....	15-404
Figure 15.5-148: All Control Rod Withdrawal at Power (ACRW) .....	15-405
Figure 15.5-149: All Control Rod Withdrawal at Power (ACRW) .....	15-406
Figure 15.5-150: All Control Rod Withdrawal at Power (ACRW) .....	15-407

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-151: All Control Rod Withdrawal at Power (ACRW) .....	15-408
Figure 15.5-152: All Control Rod Withdrawal at Power (ACRW) .....	15-409
Figure 15.5-153: All Control Rod Withdrawal at Power (ACRW) .....	15-410
Figure 15.5-154: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-411
Figure 15.5-155: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-412
Figure 15.5-156: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-413
Figure 15.5-157: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-414
Figure 15.5-158: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-415
Figure 15.5-159: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW).....	15-416
Figure 15.5-160: Feedwater Isolation (DEC) .....	15-417
Figure 15.5-161: Feedwater Isolation (DEC) .....	15-418
Figure 15.5-162: Feedwater Isolation (DEC) .....	15-419
Figure 15.5-163: Feedwater Isolation (DEC) .....	15-420
Figure 15.5-164: Feedwater Isolation (DEC) .....	15-421
Figure 15.5-165: Feedwater Isolation (DEC) .....	15-422
Figure 15.5-166: Not Used.....	15-422
Figure 15.5-167: FWLB Dose Analysis Mass Release vs Time.....	15-423
Figure 15.5-168: MCNP-06P Problem Geometry for Out of Core Criticality Model .....	15-424
Figure 15.5-169: Plot of Out of Core Criticality Dose Consequence Versus Distance.....	15-425
Figure 15.5-170: Reactor Power, Large FW Pipe Break, Conservative Case .....	15-426
Figure 15.5-171: Reactor Pressure, Large FW Pipe Break, Conservative Case .....	15-427
Figure 15.5-172: Reactor Water Level, Large FW Pipe Break, Conservative Case .....	15-428
Figure 15.5-173: Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case .....	15-429
Figure 15.5-174: Containment Pressure, Large FW Pipe Break, Conservative Case .....	15-430
Figure 15.5-175: Containment Temperature, Large FW Pipe Break, Conservative Case ...	15-431

## 15.0 SAFETY ANALYSIS

This chapter includes the following safety analyses complying with the requirements of CNSC REGDOC-1.1.2, Section 4.4.1, Draft Version 2 (Reference 15.1-1):

- Hazards Analysis
- Deterministic Safety Analysis (DSA)
- Probabilistic Safety Assessment (PSA)

The safety analysis primary objective is demonstrating that the Fundamental Safety Functions (FSFs) (see Chapter 3, Subsection 3.1.5) are effective in:

- Controlling reactivity
- Removing heat from the fuel (reactor or fuel pool)
- Confining radioactive materials
  - Shielding against radiation
  - Controlling operational discharges
  - Limiting accidental releases

The BWRX-300 design basis is achieved through an iterative safety process. The design is implemented to meet defined safety objectives that are confirmed via the safety analyses. Results of the safety analyses provide feedback to the design. If indicated by the results, the design may be modified until safety objectives are met.

This chapter documents the safety objective and the safety analyses approach, description and results performed for the BWRX-300 design. The DSA and the PSA are conducted in compliance with CNSC REGDOC-2.4.1 (Reference 15.1-2), and CNSC REGDOC-2.4.2 (Reference 15.1-3), respectively. The content of Chapter 15 is structured as follows:

Section 15.1 General Considerations of the BWRX-300 Safety Analyses:

- Safety Analysis Objectives, Scope, and Approach
- Analysis of Hazards – Scope and Approach
- Analysis of Design Basis Conditions - Scope and Approach
- Analysis of Design Extension Conditions Without Core Damage Scope and Approach
- Analysis of Beyond Design Basis Accidents (BDBAs) with Core Damage (Severe Accident)- Scope and Approach

Section 15.2 Identification, Categorization, and Grouping of Postulated Initiating Events and Accident Scenarios discusses:

- Development of categorizing events according to their frequencies
- Grouping of events according to their types
- Selection of the bounding events for each group type

Section 15.3 Safety Objectives and Acceptance Criteria discusses:

- Safety objectives based upon the safety strategy

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- Associated acceptance criteria for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and safety goals

Section 15.4 Human Actions discusses:

- General Considerations
- Human Actions considered in the DSA
- Human Actions considered in the PSA

Section 15.5 Deterministic Safety Analysis discusses:

- Safety margins in the safety analysis (Subsection 15.5.1)
- Analysis of normal operations (Subsection 15.5.2)
- Analysis of AOOs (Subsection 15.5.3)
- Analysis of DBAs (Subsection 15.5.4)
- Analysis of Design Extension Conditions (DECs) without core damage (Subsection 15.5.5)
- Analysis of DECs with core damage (Subsection 15.5.6)
- Analysis of PIEs and accident scenarios associated with the fuel pool (Subsection 15.5.7)
- Analysis of fuel handling events (Subsection 15.5.8)
- Analysis of radioactive release from a system or component outside containment Loss-of-Coolant-Accidents (LOCA) outside containment (Subsection 15.5.9)
- Analysis of internal and external hazards (Subsection 15.5.10)

Section 15.6 Probabilistic Safety Assessment discusses:

- General approach to the PSA (Subsection 15.6.1)
- Results of the Level 1 PSA (Subsection 15.6.2)
- Results of the Level 2 PSA (Subsection 15.6.3)

Section 15.7 Summary Results of the Safety Analyses discusses:

- Analysis results of normal operation (Subsection 15.7.1)
- Analysis results of AOOs and DBAs (Subsection 15.7.2)
- Analysis results of DECs without core damage (Subsection 15.7.3)
- Analysis results of DECs with core damage (Subsection 15.7.4)
- Analysis results of fuel pool events (Subsection 15.7.5)
- Analysis results of fuel handling events (Subsection 15.7.6)
- Analysis results of radioactive releases from LOCAs outside containment (Subsection 15.7.7)
- Analysis results of internal and external hazards (Subsection 15.7.8)
- Analysis results of the PSA (Subsection 15.7.9)

Appendix 15A discusses reference source term for conditions that are practically eliminated.

Appendix 15B discusses complementary design features for mitigating DEC's.

Chapter 15 documents the analysis and results from the BWRX-300 plant safety analyses.

### **15.1 General Considerations of the BWRX-300 Safety Analyses**

The BWRX-300 safety strategy framework integrates the Defense Lines (DLs) provided by the implementation of the Defence-in-Depth (D-in-D) concept (defined in CNSC REGDOC-2.5.2, Section 4.3.1 (Reference 15.1-4) with the safety analyses. The D-in-D concept uses insights gained from operating experience and deterministic and risk-informed and performance-based analyses.

Chapter 3 defines the safety strategy framework implementation process shown on Figure 3.1-2. Design features, functions, and practices are organized into DLs that protect the integrity of the physical barriers against the radioactive releases. Safety analyses are performed that demonstrate the effectiveness of the Structures, Systems, and Components (SSCs) necessary to perform the functions assigned in various DLs that are credited to mitigate the Postulated Initiating Events (PIEs). The SSCs necessary to achieve the DL safety functions are described in Chapters 4 through 8. The PIEs analyzed are identified and selected through the systematic process of fault evaluation described in Section 15.2.

The safety analyses prove that the plant design meets the underlying safety objectives and acceptance criteria for event mitigation and confirms that the applicable regulatory safety objectives are met.

#### **15.1.1 Safety Analysis Objectives, Scope, and Approach**

The general Nuclear Safety Objectives in Chapter 3, Subsection 3.1.1 are demonstrated by the results from the safety analyses. Chapter 3, Figure 3.1-2 presents a graphical representation of the safety analysis performed according to the BWRX-300 safety strategy framework that includes:

- Hazards Analysis
- Deterministic Safety Analysis (DSA)
- Probabilistic Safety Assessment (PSA)

The Hazard Analysis consists of four types of hazard evaluations described in Subsection 15.1.2. The main objectives of the hazard evaluations are the identification of potential PIEs and confirmation that the plant design effectively responds to credible internal and external hazards.

The safety analysis objectives include:

- Demonstrating the design meets the acceptance criteria established following a graded approach for each plant state. The graded approach application may lead to acceptance criteria more restrictive for events with higher occurrence probability.
- Deriving and confirming Operational Limits and Conditions (OLCs) for normal operation
- Establishing and validating accident management procedures and guidelines

The safety analysis scope includes the plant states described in Chapter 3, Subsection 3.1.3 and illustrated in Figure 3.1-1. The plant states are consistent with CNSC REGDOC-2.5.2, Section 7.3 and includes:

- Normal operation
- AOOs



- DBAs
- BDBAs

The subset of BDBA or DECAs may occur without core damage or with core damage.

The hazards analysis consists of four types of hazard evaluations described in Subsection 15.1.3. The hazards evaluation main objective is the identifying potential PIEs and demonstrating the plant design effectively responds to credible internal and external hazards.

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA evaluations:

- Baseline – deterministic safety analysis (BL-DSA)
- Conservative – DSA (CN-DSA)
- Extended – deterministic safety analysis (EX-DSA)

This approach addresses initiating and mitigating DL function failures in a more systematic and structured manner than past approaches.

These DSA acceptance criteria are discussed in Section 15.3. The DSA results are compared against the applicable plant state acceptance criteria and dose limits specified in Subsection 15.3.1.

The PSA is performed to complement the DSA. PSA estimates the overall risk presented by the facility that is compared to the regulatory safety goals specified in Subsection 15.3.2. The PSA is presented in Section 15.6.

### **15.1.2 Analysis of Hazards**

An initial step in performing the safety analysis is a systematic hazards evaluation. The BWRX-300 Safety Strategy process described in Chapter 3, Subsection 3.1.6.4 identifies four types of hazard evaluations for the complete range of operational modes (full power, low power, load following, shutdown and refueling) that produces a comprehensive set of PIEs:

- Functional Failure Hazard Evaluation (FFHE)
- External Hazard Evaluation (EHE) – addressed in Chapters 3, 6, Sections 3.3, 6.5, respectively
- Internal Hazard Evaluation (IHE) Chapter 3, Section 3.4, Chapter 6, Section 6.5, Chapter 9A, Section 9A.6
- Human Operation Hazard Evaluation (HOHE)

The hazard evaluations include any consequential failure that occurs because of the PIE. They also address all sources of radioactivity (e.g., spent fuel, fuel being handled) in addition to the reactor core itself.

Each hazard evaluation identifies any potential challenges to an FSF.

#### **15.1.2.1 Functional Failure Hazard Evaluation**

The FFHE identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. These hazards are identified in Failure Modes and Effect Analyses (FMEAs) performed on the plant systems.

The FFHE is limited to random single failures and to CCFs. The system FMEAs are reviewed to identify failures that cause challenges to FSFs. A consolidated list of failures from all system FMEAs is generated and organized.

The functional failure hazard potential PIE sources are organized by quantitative frequency, using the frequency ranges defined in the Safety Strategy.

#### **15.1.2.2 External Hazard Evaluation**

The EHE includes natural and human-induced hazards that originate from a source that is not under control of the nuclear power plant licence holder. The EHE addresses individual hazard sources and combinations of sources:

- Natural external hazards include earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions
- Human-induced external hazards include toxic gas releases, aircraft crashes, or ship collisions

External events are site-specific and are specified in the site evaluation provided in Chapter 2.

Once the external hazards are identified, the BWRX-300 structures are designed to withstand these external hazards, and the resulting protection is described in Chapter 9B. Human-induced external hazards such as toxic gas and aircraft impacts are provided in Chapter 6, Section 6.5 for control room habitability.

The sources of external hazard or combinations are organized by quantitative frequency as potential PIEs evaluated in the fault evaluation (see Chapter 3, Section 3.3). Malevolent acts are addressed in the Security Annex.

#### **15.1.2.3 Internal Hazard Evaluation**

The IHE identifies conditions originating within the boundaries of the site and with potential to lead to an unplanned plant transient. The internal hazard condition does not directly challenge an FSF, but the effects of the hazard may cause equipment failures. These equipment failures are then evaluated in the deterministic and probabilistic safety analyses.

Internal hazards include:

- Fires (discussed in Chapter 9A, Section 9A.6)
- Explosions, missiles from rotating or pressurized equipment (discussed in Chapter 3, Section 3.4)
- Collapse of structures/falling objects (discussed in Chapter 9B)
- Pipe whip, jet effects, and flooding (discussed in Chapter 3, Section 3.6)

The IHE addresses both individual hazard sources and combinations of sources.

The sources of internal hazard or combinations are organized by quantitative frequency as potential PIEs and evaluated in the fault evaluation (see Chapter 3, Section 3.4).

#### **15.1.2.4 Human Operation Hazard Evaluation**

The HOHE identifies erroneous decisions or human action(s) that lead to an unplanned plant transient. Human operations hazards, typically involve unplanned changes to plant equipment status by equipment operators or maintenance personnel.

Many human operations hazards produce the same effects as corresponding equipment failures and the effects of these are included in the FFHE described in Subsection 15.1.3.1. The HOHE

is limited to a single erroneous act that may lead to multiple system responses. The HOHE focuses on identifying unique hazards such as an operator initiating a group command on multiple actuators that is beyond what is considered in a single failure analysis of a particular system.

The sources of HOHE or combinations are organized by quantitative frequency as potential PIEs and evaluated in the fault evaluation (see Section 15.5).

### 15.1.3 Analysis of Design Basis Conditions

The BWRX-300 design basis conditions are normal operations, AOOs and DBAs described below:

1. **Normal Operation** is operation within specified OLCs (see Chapter 16) and includes the full range of plant operating modes (Chapter 1, Section 1.8.). The objective of the normal operation safety analysis is to demonstrate that DL1 measures are effective in preventing abnormal operations and failures, thus meeting radiological requirements.
2. **AOOs** are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility. The objective of the AOO safety analysis is to demonstrate that DL2 functions are effective for most AOO PIEs in meeting the applicable acceptance criteria.
3. **DBAs** conditions are identified as deviations from normal operations that are less frequent and more severe than AOOs. An objective of DBA safety analysis is to demonstrate that DL3 functions are effective in mitigating events and meeting the applicable acceptance criteria.

Acceptance criteria applicable to the DSA for each plant state is discussed in Subsection 15.3.1. The response to AOOs and DBAs is achieved by SSCs specifically designed to mitigate these events and are assigned DL2 and DL3 functions (Chapter 3, Subsection 3.1.6.2).

The DSA results for design basis conditions in Section 15.7 demonstrate that the requirements of CNSC REGDOC-2.4.1 are met.

### 15.1.4 Analysis of Design Extension Conditions Without Core Damage

DECs are a subset of BDBAs. DECs are postulated accident conditions that are less frequent than DBAs. DECs may occur with or without core damage.

DSA is performed for DECs without core damage demonstrating that releases of radioactive material are kept within acceptable limits and support the PSA determination of no core damage.

DEC analysis include:

- Multiple failures defined as complex sequences identified in the Level 1 PSA or as a PIE with a Common Cause Failure (CCF)
- AOO and DBAs with postulated failures of DL2 and DL3 functions analyzed in EX-DSA. For these events, the DBA acceptance criteria are used as screening criteria to the evaluation of core damage
- Low frequency events
- Non-reactor fault sequences (fuel pool accidents) are analyzed in Level 1 PSA

The results of the DSA for DECs without core damage are discussed in Section 15.7. The analysis of DECs with core damage are addressed in the Level 2 PSA described in Subsection 15.6.4.

#### **15.1.5 Analysis of Beyond Design Basis Accidents with Core Damage (Severe Accident)**

These are referred to as Severe Accidents (SA) and involve a catastrophic failure, core damage, and fission product release. A SA is generally considered to begin with the onset of core damage. To the extent that core damage is not practically eliminated, representative DECAs with core damage are postulated to provide inputs for the containment design and safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECAs and represents a set of bounding cases. Accident scenarios considered for practical elimination are described in Appendix 15A.

Severe accident sequences are selected that identify representative core damage scenarios and corresponding plant damage states that are used as the basis for performing the Severe Accident Analysis (SAA). The scope of SA scenario selection corresponds to sequences involving significant core damage that could lead to a containment breach and radioactive release analyzed in the Level 2 PSA in Section 15.6. The selected SA scenarios are included in a fault evaluation.

The SAA goal is to provide input to accident management for terminating the progression of core damage, maintaining containment integrity as long as possible, and minimizing on-site and offsite radioactive material releases. Halting core damage progress prevents Reactor Pressure Vessel (RPV) failure.

The response to SAs considers the use of safety and non-safety, permanent and temporary systems and equipment that are beyond their originally intended functions.

Consistent with SSR 2/1, Paragraph 2.11 (Reference 15.3-3), practical elimination is applied to events or sequences of events leading to or involving core damage (a severe accident) where confinement of radioactive materials cannot be reasonably achieved. Event sequences that are either physically impossible or extremely unlikely to occur are considered for practical elimination.

The practical elimination demonstration is performed with accident conditions and phenomena knowledge and is substantiated by relevant evidence (see Appendix 15A Reference Source Term for Conditions That Are Practically Eliminated).

#### **15.1.6 References**

- 15.1-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 15.1-2 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 15.1-3 CNSC Regulatory Document REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants," May 2014.
- 15.1-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

## **15.2 Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios**

A fundamental element of the safety analyses is the identification and selection of PIEs that is achieved through systematic process of fault evaluation. The fault evaluation objective includes:

- Identification, categorization and grouping of PIEs
- Identification of the plant functions expected to be credited in the safety analysis and their assignment to a functional DL (DL2, DL3 and DL4)

The fault evaluation scope is the list of potential PIEs generated by the hazards evaluation (see Subsection 15.1.3) and includes:

- Complete range of operating modes
- All radioactivity sources (reactor core and outside core)
- Single failure PIEs, CCF PIEs, and fault sequences developed based on the success or failure of mitigating functions. Single and CCF PIEs include equipment failures and human errors. CCF are analyzed as software platform or mechanical failures.

The output of the fault evaluation is documented in a fault list. The fault list establishes traceability between the plant design and the safety analysis. The fault evaluations start in parallel with or prior to DSA and PSA activities. DSA and PSA mature with the design and the fault list is updated accordingly.

The fault evaluation process used for PIE identification and selection complies with CNSC REGDOC-2.4.1, Section 4.2, "Events to be analyzed" (Reference 15.3-1).

The fault evaluation includes the following activities as shown in Chapter 3, Figure 3.1-2 BWRX-300 Safety Strategy Implementation Process.

- Deterministic PIE Selection
- Complex Sequence Selection
- Severe Accident Scenario Selection

### **Deterministic PIE Selection**

The deterministic PIE selection is the systematic process in organizing and selecting events for deterministic safety analyses. The activities performed during the deterministic PIE selection are described in Subsection 15.2.1. Selected PIEs and fault sequences are allocated to three DSA types in a fault list:

- PIE List for Baseline Deterministic Safety Analysis (BL-DSA)
- PIE/Fault Sequence List for Conservative Deterministic Safety Analysis (CN-DSA)
- PIE/Fault Sequence List for Extended Deterministic Safety Analysis (EX-DSA)

### **Complex Sequence Selection**

Complex sequences are fault sequences involving failures of multiple mitigating features, which have not been included in the deterministic PIE selection but are identified in the Level 1 PSA as having the potential to lead to core damage with a frequency of occurrence or consequences judged to require analysis and DL mitigation function. These complex sequences are added to the fault list and analyzed in the EX-DSA (Subsection 15.1.5).

## **Severe Accident Scenario Selection**

The scope of severe accident sequence selection corresponds to those sequences involving significant core damage, which could lead to a breach of containment and radioactive release in the Level 2 PSA that is described in Section 15.6. To the extent that core damage is not practically eliminated, representative severe accident sequences (DECs with core damage) are postulated and analyzed in the SAA. The primary objective of the severe accident sequence selection is to identify representative core damage scenarios and define corresponding plant damage states that are used as the basis for performing the SAA. The selected SA scenarios are documented in the fault list and are analyzed in the SAA (Subsection 15.1.6).

### **15.2.1 Basis for Categorization of Postulated Initiating Events, Accident Scenarios and Fault Evaluation**

The hazard evaluations described in Subsection 15.1.3 results in the list of potential PIEs. These potential PIEs are evaluated, categorized, and grouped during the fault evaluation in the deterministic PIE selection. The hazard evaluations address a complete range of plant modes of operation, all sources of radioactivity and any consequential failure that occurs because of the PIE. During design development, the hazard evaluation is validated, and PIE selection is updated accordingly.

The activities included in the deterministic PIE selection and their bases are presented below:

1. Fault Sequences development – a Fault Sequence is developed starting with a PIE and considers the success or failure of the required mitigating functions. The DL of each credited mitigating function is established for each Fault Sequence.
2. Fault sequences are grouped into fault groups based on similar impact on a certain plant parameter: for example, events that lead to pressure increase in the reactor such as inadvertent closure of the Turbine Stop Valves and/or Turbine Control Valves or inadvertent closure of the Main Steam Reactor Isolation Valve(s) (MSRIVs) are grouped in the pressure increase fault group. Subsection 15.2.3 includes the output of the grouping activity.
3. Fault sequences are categorized within each fault group as AOO, DBA or DEC based on their frequency of occurrence. Subsection 15.2.2 describes this categorization.
4. Plant conditions are defined corresponding to each PIE supporting the scenario analysis.
5. Any exceptions are applied or justified to the standard PIE selection.

Fault sequences are allocated to three types of DSAs:

- BL-AOO and Baseline Design Basis Accident (BL-DBA) – Baseline DSA
- CN-AOO and CN-DBA –Conservative DSA
- EX-DEC –Extended DSA

A bounding set of PIEs and fault sequences that result in the most significant challenge to the FSFs are selected for evaluation in the DSA. DSA layers and events categories are combined so that limiting baseline events are AOO (BL-AOO), the limiting CN events are DBAs (CN-DBA) and limiting EX events are DECs (EX-DEC). This notation is used to identify the layer and event category.

Subsection 15.2.4 describes the bounding event selection for each fault group that is captured in a fault list. In addition, the fault list includes the complex sequences and the severe accident scenarios specified in Subsection 15.2.4.

A description of the three deterministic safety analyses aligned with the functional DLs (DL2, DL3 and DL4a) is included below.

#### **15.2.1.1 Baseline Deterministic Safety Analysis**

The primary objective of the BL-DSA is demonstrating the effectiveness of the DL2 functions. The scope of BL-DSA includes single failure PIEs categorized as bounding BL-AOOs and BL-DBAs. The BL-DSA models the expected response of the plant (no failure is postulated) to demonstrate that the event meets applicable acceptance criteria. The analysis end point is the controlled state condition. The mitigating DL functions credited in BL-DSA are DL2 functions. If a DL2 function fails or is not effective, then the corresponding DL3 function is credited.

#### **15.2.1.2 Conservative Deterministic Safety Analysis**

CN-DSA primary objective is demonstrating the effectiveness of DL3 functions. The CN-DSA scope includes events categorized as bounding CN-AOOs and CN-DBAs:

- PIEs due to single failure
- PIEs due to spurious CCF in DL2 or DL4a
- Baseline PIEs with postulated passive CCF of DL2 functions that were credited in BL-DSA

The CN-DSA is performed using conservative initial conditions with established acceptance criteria and applying a graded approach in quantifying the uncertainties (see Subsection 15.5.1.1). Single failure criterion is applied to DL3 SC1 SSCs. CN-DSA credits only DL3 mitigation functions. The end point of the analysis is a controlled state condition.

#### **15.2.1.3 Extended Deterministic Safety Analyses**

The EX-DSA primary objective is assessing the effectiveness of DL4 functions. The EX-DSA scope includes events categorized as DEC:

- PIEs due to spurious CCF of DL3 functions
- DBA fault sequences with postulated passive CCF of DL3 mitigating functions
- Complex sequences identified by the Level 1 PSA

An extended sequence for AOOs and DBAs is required in the following conditions:

1. If a DL3 function is credited to mitigate a single failure PIE in the BL-DSA, then the DEC fault sequence assumes a passive CCF in DL3 functions (no additional mitigation single failure is assumed).
2. If the hydraulic scram action is credited in an BL-AOO scenario, then the hydraulic scram action is assumed to have a mechanical CCF of the hydraulic scram where only the Control Rod Drive Motor (CRDM) run-in functions insert control rods. No additional failures are assumed.

#### **15.2.2 Categorization of Events According to Their Frequencies**

One fundamental element of the deterministic PIE selection (Section 15.1) and fault sequence selection is the assignment of fault sequences to categories based on their frequency of occurrence that complies with CNSC REGDOC-2.4.1, Sections 4.2.2.5 and 4.2.3, and CNSC REGDOC-2.5.2, Section 5.4.3:

- Anticipated Operational Occurrence (frequency greater than 1E-02 per reactor-year)
- Design Basis Accident (frequency between 1E-02 and 1E-05 per reactor-year)

- Design Extension Condition (frequency less than 1E-05 per reactor-year)

Qualitative frequencies are adopted as an interim measure and are used in the early design stages to progress the performance of deterministic analyses prior to availability of more mature PSA information. Quantitative frequencies based on Level 1 PSA results are adopted as the final, governing measure of the event sequence category.

A fault sequence consists of a combination of a PIE and can include an assumed failure of a mitigating function(s). The event sequence category is based on the sequence frequency not only the PIE frequency. The event category assigned to an event sequence may be different than the event category assigned to the PIE that initiated the sequence because the event sequence may include additional failures that make the sequence less likely to occur.

In addition to the event categorization frequency, the categorized events are allocated the following DSA types:

- Baseline Anticipated Operational Occurrence (BL-AOO)
- Conservative Anticipated Operational Occurrence (CN-AOO)
- Baseline Design Basis Accident (BL-DBA)
- Conservative Design Basis Accident (CN-DBA)
- Extended Design Extension Condition (EX-DEC)

### **15.2.3 Grouping of Events According to Type**

One of the steps in fault evaluation is grouping the events according to their type. The fault evaluation includes external events, internal events, human operational errors, functional failures evaluated in hazard analysis, Level 1 PSA complex sequences and Level 2 PSA severe accident sequences.

PIEs (faults) are grouped according to the resultant change in plant parameter:

- Temperature decrease events – decrease in core coolant temperature
- Pressure increase events – increase in reactor pressure
- Reactivity increase events – reactivity and power distribution anomalies
- Inventory increase events – increase in reactor coolant inventory
- Inventory reduction events – decrease in reactor coolant inventory
- Non-reactor fault events – these events are non-core related such as fuel handling accident
- Radiological faults having dose consequences

Once the fault groups are identified, then the anticipated core physics response associated with each group is then selected. Once each group is identified, then the bounding fault sequence from that group is selected.

Table 15.2-1 provides the fault groups with an explanation of how anticipated core physics response (reactor response) was considered in development of the groups. Within these groups, fault sequences with similar responses are compared and used to select the bounding events.



#### **15.2.3.1 Core Reactivity Effects of PIEs and Accident Scenarios**

The anticipatory effects of core reactivity response provided in Table 15.2-1 are discussed below with focus on AOOs, DBAs, and DECAs without core damage because these are event categories where the core remains intact, and the reactivity feedback mechanisms are well understood.

##### **Void Reactivity**

Void reactivity is an important reactivity feedback mechanism in BWR transient and accident analyses. The void reactivity feedback is always negative and is typically stronger (more negative void coefficient) at the end of an operating cycle. It is also stronger for reload cores versus an initial reactor core that includes only fresh fuel. This reactivity feedback mechanism is the dominant feedback for some PIE groups. Table 15.2-1 focuses on events initiated from conditions of normal power operation. In the DSA, where it is important, void reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response analyzed using Transient Reactor Analysis Code General Electric (TRACG), the primary DSA computer code (see Subsection 15.5.1.2).

##### **Moderator Temperature Reactivity**

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature. The value of this coefficient is important during the startup of a BWR. During power operation, the coefficient is not important, because the moderator is boiling and remains at the saturation temperature corresponding to the operating pressure. In addition, moderator density changes caused by boiling are much larger than changes from moderator temperature changes and therefore, mask any effects. The BWRX-300 core is designed and evaluated to conform to regulatory requirements as discussed in Chapter 4, Sections 4.2 - 4.4. Moderator temperature feedback is accounted for in the DSA analysis of startup conditions when coolant temperature is key to the event response and there is no significant voiding in the core. Once core boiling/voiding begins, void reactivity feedback becomes dominant.

##### **Control Reactivity**

Neutron absorbing control rods are the primary means to control reactivity in transient and accident analyses. During events that result in relatively fast positive reactivity feedback, control rods are inserted rapidly using stored hydraulic energy. This is referred to as a “scram”. During events that result in relatively slow positive reactivity feedback and do not require a reactor “scram”, the Fine Motion Control Rod Drive (FMC RDs) that are operated using electric motors can be used for slower control rod insertion, (see Chapter 4, Subsection 4.5.1). Reactivity feedback mechanism is key for some PIEs groups (see Table 15.2-1) in which rods are withdrawn in error. Control reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

##### **Doppler Reactivity**

Doppler reactivity is a less important reactivity feedback mechanism in BWR transient and accident analyses than void or control reactivity. Doppler reactivity is negative with an increase in fuel temperature and becomes more important as the fuel temperature continues to increase. In DSA where reactivity feedback is important, doppler reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

##### **Boron Reactivity**

For the BWRX-300, boron reactivity insertion (see Appendix 15B) is only needed for long-term shutdown in very low probability events where the hydraulic and electric motors fail to insert a sufficient number of control rods.

## **Xenon Reactivity**

Xenon reactivity feedback is not typically accounted for during events in DSA because the rate of change of reactivity is slow. The effects of xenon are accounted for in analyses of shutdown margin.

### **15.2.4 Postulated Initiating Events and Accident Scenarios**

PIEs and event frequency are first determined qualitatively based on system conceptual design, previous similar designs, and operating experience. PIEs are evaluated in the fault evaluation (see Section 15.2) where they are further screened for inclusion in the fault list.

The bounding event selection is performed for events that are initiated at full power conditions (Mode 1 operating condition) because they are expected to result in the most significant challenge to the fission product barriers.

Bounding events are selected in each fault group, for each event category (e.g., AOO, DBA, DEC without core damage) and for the applicable DSA layer (e.g., baseline, conservative and extended). The resulting events selected are listed in Table 15.2-2 and analyzed in Section 15.5. DEC events with core damage are part of the PSA and SAA.

The bounding event selection is performed for two event categories:

- Transient or non-LOCA described in Sections 15.2.4.1 through 15.2.4.6
- LOCA scenarios described in Section 15.2.4.7

### **Bounding Event Selection for Transient Events**

Bounding events are selected for the transient (non-LOCA) DSA that pose the most challenges in meeting the derived acceptance criteria.

The selected bounding events are summarized in Table 15.2-2. Table 15.2-2 also points to a complete description of the bounding event in the DSA described in Section 15.5.

#### **15.2.4.1 Decrease in Core Coolant Temperature Bounding Event**

Events that result in core coolant temperature decreases are grouped as Temperature Decrease (TD) faults. A reduction in coolant temperature (at the core inlet) has the potential to challenge the fuel cladding barrier due to increasing reactivity as a result of reducing the core coolant void fraction. The Reactor Coolant Pressure Boundary (RCPB) is not challenged because there is not a significant increase in steam flow and normal pressure control is not affected. Because this is a reactivity driven event, it is only of concern when the reactor is not shutdown.

The largest source of coolant supply is from the FW pumps. FW flow enters the RPV downcomer through the FW piping. Extraction steam from the turbine is directed to heat FW in multiple stages of FW heaters. Failures, such as loss of extraction steam, can result in a reduction of the temperature entering the RPV.

The Isolation Condenser System (ICS), Control Rod Drive System (CRD), Reactor Water Cleanup System (CUW) and the Shutdown Cooling System (SDC) also have inflows to or outflows from the RPV that have the potential to reduce the coolant temperature. Inadvertent ICS initiation PIEs are included in the increase in reactor coolant inventory fault group. These systems can only reduce the coolant temperature a small fraction relative to Feedwater Heating (FWH)-related PIEs. Therefore, only FW heater related PIEs are considered potentially bounding.

There is no fault group for increase in core coolant temperature for the following reasons:

1. The Feedwater (FW) temperature is near the highest temperature that is feasible during normal operation.
2. Any increase in FW temperature would increase the core void fraction and reduce core power due to the decrease in void reactivity.
3. An increase in FW temperature does not result in an increase in core temperature because the core is boiling and remains at saturated conditions.

Other than an increase in the FW temperature, the core coolant temperature may increase due to the following conditions:

- Core power increase resulting in a small increase in pressure covered by the reactivity and power distribution anomalies fault group
- Reactor pressure increase that effects saturation temperature covered by increase in reactor pressure fault groups

Therefore, any possible increase in FW temperature is small and results in a small decrease in power and poses no threat to fission product barriers resulting in no need for an increase in core coolant temperature fault group.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

#### **Bounding AOO Event Selection**

The event effect on the fuel Critical Power Ratio (CPR) and core power increase is used in selecting the bounding AOO events for this group.

The AOO that results in the largest reduction in FW temperature is the Loss of Feedwater Heating (LFWH) event. This BL-AOO event described in Subsection 15.5.3.1.1 is the bounding AOO event for this category.

#### **Bounding DBA Event Selection**

The bounding CN-DBA event described in Subsection 15.5.4.1.1 resulting in the largest postulated reduction in FW temperature is the CCF leading to loss of all FW heaters. The event effect on fuel Peak Cladding Temperature (PCT) is used in selecting the bounding DBA in this group.

#### **Bounding DEC Without Core Damage Event Selection**

There are no DEC events in this fault group because the DLs are established in the baseline and conservative DSA. No complex sequences in the fault group are identified.

#### **15.2.4.2 Increase in Reactor Pressure Bounding Event Selection**

Events that result in RCPB pressure increase are referred to as Pressure Increase (PI) faults. During full power operation, steam generated in the reactor exits the RPV through the Main Steam Lines (MSLs). There are normally open valves in the each MSL: Two Main Steam Reactor Isolation Valves (MSRIV) and one main steam Containment Isolation Valve (CIV). Outside containment, there is a main steam header upstream of the Turbine Stop Valves (TSVs). Downstream of the main steam header the TSVs are in series with the Turbine Control Valves (TCVs).

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The MSRIVs and CIVs close to isolate the RPV or containment. The TSVs and TCVs close to protect the turbine. Turbine Bypass Valves (TBVs) located in the main steam equalizing line in the Turbine Building (TB) allow steam to bypass the closed TSV or TCV.

Pressure increase faults are generally caused by closure of valves in the steam flow path. Closure of a single MSRIV terminates steam flow in one of the two MSLs. Closure of any two of these valves in separate MSLs causes a complete isolation of the MSLs upstream of the TBVs. Closure of both TSVs and/or TCVs, terminates steam flow to the turbine. A turbine trip or load rejection signal initiates closure of the TSVs and TCVs, and fast opening of the TBVs. When RPV pressure increases, TBVs are opened by the normally operating Reactor Pressure Control (RPC) which allows up to ~25% of the rated steam flow to the main condenser.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

There is no fault group for decrease in reactor pressure because a pressure decrease reduces reactivity. The decrease in reactor pressure is caused by either:

- Reactor power decrease that is covered by the reactivity at power distribution anomalies fault group
- Reactor coolant inventory is lost that is covered by the decrease in reactor coolant inventory fault groups.

Therefore, there is no need for a decrease in reactor pressure fault group.

#### **Bounding AOO Event Selection**

The event effect on the fuel CPR, core power increase, and the RCPB pressure are used in selecting the bounding AOO events for this group.

There are several AOO events in the PI group in the fault list. Most of the AOOs are PIEs that result in closure of the TSVs, TCVs or both. These events present similar challenge to the cladding and RCPB, and all of them are selected as potentially limiting (bounding) events:

- Turbine trip, load rejection
- Loss-of-preferred power
- Loss of condenser vacuum

Another AOO in this group is the BL-AOO closure of one Main Steam Reactor Isolation Valve (MSRIVC). The MSRIVs do not close as fast as the TCVs or TSVs; however, the MSRIVs are upstream of the TBVs, resulting in this event selected as a potentially limiting.

#### **Bounding DBA Event Selection**

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DBA events for this group.

There are several DBA events that result in increased pressure in the PI group. There are two main types assumed in the PI group:

1. AOO events where the TSV or TCV closes with a CCF of the DL2 mitigation equipment
2. CCF results in closure of the MSRIV and FWRIV

Because the TCV fast closure is faster than the MSRIV closure, the closure of the TCVs with failure of the DL2 mitigation functions is a more significant pressure increase event. However,

because several of these events result in similar challenges to acceptance criteria, several CN-DBA events (described in Subsection 15.5.4) are chosen as potentially bounding.

### **Bounding DEC Without Core Damage Event Selection**

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DEC events for this group.

The main types of PI group DEC events are:

1. Bounding AOO events with failure of the hydraulic scram due to postulated CCF of the hydraulic components
2. Complex sequence AOO event with half of the control rods fail to insert either by HCUs or with the CRD motors run-in
3. CCF in DL3 functions initiating events

The events assume a CCF of the hydraulic scram that significantly challenges the fuel cladding and RCPB because the CRD motor run-in that backs up the hydraulic scram results in slower negative reactivity insertion. Several of these events are selected as potentially bounding. The event result from CCF in DL3 functions are not concerning because sufficient non-DL3 functions initiate the hydraulic scram.

In the TSV and TCV closure events, pressure control is achieved (at least momentarily) via the TBVs to the main condenser. In the loss of condenser vacuum and LOPP AOO events with failure of the hydraulic scram, the main condenser is available for a limited amount of time. These events result in more severe pressurization and are selected as potentially bounding relative to the turbine trip and load rejection events. The 1MSRIVC AOO with failure of the hydraulic scram is also selected.

A complex sequence is also considered in this fault group based on the PSA evaluation. This event is selected as a potentially limiting event.

### **15.2.4.3 Reactivity and Power Distribution Anomalies Bounding Event Selection**

Events that result in reactivity and power distribution anomalies are grouped as Reactivity Increase (RI) faults. These events include failures in reactivity control that challenge the fuel cladding or RCPB integrity. The BWRX-300 controls reactivity with control rod movement. Normal control is accomplished using FMCRDs, which are individually connected to the control rods. HCUs are used to quickly shutdown the reactor by inserting control rods, ensuring reactivity control during off normal operation. Increases in reactivity can result from control rod withdrawal, an error in fuel loading, or control rod drop.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

### **Bounding AOO Event Selection**

The fault list includes only one RI-AOO. This event is Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW). However, this event is not evaluated because a protection function of Automatic Thermal Limits Monitor (ATLM) initiates a control rod block. The event conditions remain within normal operation conditions; therefore, no analysis is needed. The design description of the ATLM is found in Chapter 7, Subsection 7.3.3.2.

### **Bounding DBA Event Selection**

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding events for this group.

There are several DBA PIEs in the RI group to postulate in identifying the bounding event including the withdrawal of single or multiple control rods. The DBA PIEs are all mitigated by active DL2 functions ATLM and Multi-channel Rod Block Monitor (MRBM) that result in minimal change to core power in the baseline sequence. The ATLM blocks rod motion before departure from normal operation and MRBM blocks rod motion before significant fuel effects occur. These mitigation functions are reliable resulting in a failure sequence in the DEC frequency range. These events with the failure of ATLM and MRBM are selected as bounding DEC without core damage events because of the low frequency of the sequence. The following fault sequences are evaluated.

1. CCF - All Control Rod Insertion at Power - (CCF-ACRI): For some core conditions, this event may result in a momentary increase in reactivity in the top part of the core because higher density water moves up the core faster than the rod motion. As the rods insert, high density coolant increases the reactivity above the control rods. This is a momentary effect, does not challenge fuel cladding criteria or any other fission product barrier, and is not selected as a bounding event.
2. Fuel Loading Error (FLE): This event postulates that two fuel assemblies are swapped during refueling or a single assembly is inserted in a rotated position (180 degrees from normal). This can result in small local increase in reactivity. This event is selected as a potentially limiting DBA event.
3. Control Rod Drop Accident: (CRDA): This event postulates a fault that allows separation of the control rod from the drive mechanism (design precludes a separation such as this due to a separation detection device). The control rod becomes stuck and remains in its position when the drive mechanism is attempted to be withdrawn. Before the stuck rod is detected and is inserted, the control rod becomes unstuck and falls. This event is not a limiting RI condition and not explicitly analyzed because the design includes separation detection devices that limits the drop to a small distance, resulting in no significant reactivity changes. This event is listed in Mode 2 (startup conditions) because it is limiting at those conditions. However, the event is possible at normal full power operations.

### **Bounding DEC Without Core Damage Event Selection**

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding DEC events for this group. There are two control rod withdrawal error DEC events that are potentially limiting:

- Single rod withdrawal
- Inadvertent withdrawal of all rod groups

These DEC without core damage are compared to applicable acceptance criteria demonstrating that features to prevent core damage are adequate and no further complementary design features are required.

#### **15.2.4.4 Increase in Reactor Coolant Inventory Bounding Event Selection**

Events resulting in an increase in reactor coolant inventory are included as Inventory Increase (II) faults. These events may occur from a FW Inventory Increase. Inadvertent ICS initiation scenarios do not fit well into any fault group and are included in this increase in inventory event category. Inadvertent ICS initiation results in much less Inventory Increase than a FW increase. However, the dynamics of ICS flow into the chimney is different than an increase in FW flow. These scenarios are selected for analysis.

Other systems such as CRD, CUW, SDC or Boron Injection System (BIS) may result in inventory increases. However, these systems have low flow rates compared to FW and are not considered as bounding.

The events selected result in the most significant challenge to fission product barrier in this fault group and the selected events are summarized in Table 15.2-2.

#### **Bounding AOO Event Selection**

There is one BL-AOO event, Inadvertent Isolation Condenser Initiation – One Train, and it is selected for evaluation.

#### **Bounding DBA Event Selection**

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding events in this group.

Two CN-DBA events are selected:

- Inadvertent injection of all ICS trains (bounds one train) (Inadvertent Isolation Condenser Initiation – All Trains (CCF-IIICI))
- Feedwater Flow Increase - All Pumps (CCF-FWFI). This event bounds the CN-DBA event for increase in flow of one FW pump.

#### **Bounding DEC Without Core Damage Event Selection**

There are no DEC events identified in the II fault group because for each single failure AOO or DBA PIE, there are two DLs established that mitigate the event. No AOO events credit the hydraulic scram. For all CCF PIEs, a DL is established.

#### **15.2.4.5 Decrease in Reactor Coolant Inventory Bounding Event Selection**

Events that result in a decrease in reactor coolant inventory are included as Inventory Reduction (IR) events. These events may occur from failures that result in:

- Reduction or loss in FW flow
- Opening of TBVs
- Pipe breaks (LOCA events) (discussed in Subsection 15.2.4.7)
- Misalignment of systems connected to the RPV
- Loss-of-preferred power (this event is included in the PI group)

The IR bounding events selection only includes non-LOCA events. In this fault group, maintaining inventory above the top active fuel ensures fuel cladding integrity. For non-LOCA events, there is no significant challenge to the RCPB or containment.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

#### **Bounding AOO Event Selection**

The event effect on coolant inventory is the parameter used in selecting the bounding AOO events in this group. There are two BL-AOO events in this group:

- FW pump trip - one pump (FWPT)
- Inadvertent opening of one TBV

The FWPT event is selected as potentially limiting because the TBV opening represents less inventory loss than the FW pump trip and FW makes up inventory loss during the TBV opening.

### **Bounding DBA Event Selection**

The event effect on the coolant inventory are the parameters used in selecting the bounding DBA events for this group.

There are several PIEs events in the IR group:

- Loss of all FW flow
- Opening of all TCVs and TBVs
- RPV Pressure Control Open

The selected bounding event is Loss of FW flow (LOFW) CN-DBA event. FW may also be lost by a FW isolation valve closure DL4a CCF (BL-DBA) but is not selected because this is less severe than LOFW. This is also less frequent and is categorized as a DEC when combined with a CCF of DL2 mitigation. The RPV Pressure Control Open (CCF-RPCO) is selected as potentially limiting for events where inventory is lost via the main steam line.

LOCA events bound these events because they result in a more significant challenge to the RPV inventory (fuel cooling and long-term cooling) and challenge containment temperature and pressure. The non-LOCA DBA events are not bounding for this IR fault group.

### **Bounding DEC Without Core Damage Event Selection**

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding DEC events for this group.

The FW Isolation EX-DEC event is selected as a potentially limiting event; however, it is not expected to be a significantly different response than the CN-DBA event. LOCA events bound this event, and the non-LOCA events are not bounding for this IR fault group.

#### **15.2.4.6 Bounding Event Selection for LOCA Scenarios**

The initiating events involving pipe breaks, scram, and trip initiation are identified in the fault list for Baseline (BL), Conservative (CN) and Design Extension Conditions (DECs). The evaluation scenarios are then selected to bound groups of pipe breaks. The consequences of postulated LOCAs are analyzed for fuel cladding and containment integrity and shown to meet the design basis acceptance criteria in Table 15.3-2. CN-DSA LOCA analyses demonstrate that the reactor level does not decrease below the Top of Active Fuel (TAF) or the fuel cladding does not exceed the normal operating temperatures. Meeting these acceptance criteria assures the fuel cooling acceptance criteria in Table 15.3-2 are met. The CN-DSA LOCA analyses also demonstrate that the containment acceptance criteria in Table 15.3-2 are met.

The bounding scenarios for pipe breaks fall into two categories:

- Large breaks inside or outside containment
- Small breaks inside or outside containment

The breaks in each category may be in steam pipes or liquid pipes. CN-DSA sequences are mitigated by DL3 functions alone, assuming a CCF of DL2 functions. EX-DSA fault sequences (DECs) are mitigated by DL2 and DL4a functions (see Table 15.5-50 DL2 and DL4a functions credited).

Pipe breaks attached to the RPV may be as large as the complete rupture of the largest steam or feedwater pipes, or as small as leaks in smaller pipes attached to the RPV. Although the



frequency of the largest breaks is lower than  $1\text{E-}05/\text{year}$ , all sizes of pipe breaks resulting from various PIEs are conservatively analyzed as DBAs. Further, the largest breaks are instantaneous double-ended guillotine ruptures of the large pipes. This assumption is made to bound the thermal hydraulic response of the fuel, RPV and containment for all isolatable pipe breaks.

The bounding LOCA events provided in Table 15.2-2 and evaluated in Subsections 15.5.4.6 (LOCA inside containment) and 15.5.9.2 (LOCA outside containment) provide the assumptions and DL components used in mitigating the event with the corresponding signals, times, and other design parameters.

A break between the RPV and the Reactor Isolation Valves (RIV) is not a credible postulated accident. Breaks inside containment are postulated to occur at any arbitrary location between the outer RIV, or the flow limiter for MS pipes, and the containment boundary.

### **Large Breaks**

Pipes that are larger than 19 mm (0.75 in) inside diameter have two isolation valves attached directly to the RPV. The largest postulated pipe breaks are in the main steam, feedwater, and ICS lines. These pipes have RIVs, which close in less than 5 seconds once they start closing. Another 5-second delay is assumed before a RIV starts closing to account for delays in break detection and signal development. A break between the RPV and RIVs is not credible as a postulated accident in the current fault list.

MS pipes are also equipped with a flow limiter to prevent very large break flow prior to MS RIV closure. The flow limiters are placed close to the RPV. Breaks on the MS pipes are postulated to occur downstream of the flow limiter.

The analytical limit for the high containment pressure setpoint is reached within 1 second for large breaks (steam and liquid). Peak containment pressure occurs at approximately the time RIVs are fully closed and the break is isolated. The largest steam pipe breaks and the largest liquid pipe breaks are the most limiting for containment response because they have the highest mass and energy release until the break is isolated. For medium size breaks that are isolated on high containment pressure, there may be a delay in reaching the containment high pressure setpoint for isolation. However, containment pressure is increasing at a slower rate than it would for a larger break while the RIVs are closing. Because the RIV closure time is the same for all breaks, the containment peak pressure is smaller for a medium size break than it is for a large break.

Since the largest breaks are more limiting for the mass and energy release, fuel integrity and containment integrity, a break spectrum analysis is not required for isolatable breaks. This is also the case for breaks outside containment since they are isolated by the leak detection system in a similar manner.

The leakage detection system is designed to detect breaks in large pipes connected to the RPV. If the break is not detected, a conservative assumption is that the break remains unisolated since no operator action is credited.

For large break LOCAs, RIVs close rapidly and prevent significant loss of RPV inventory. The core remains covered throughout. The remainder of the event after the RIVs are closed is an isolation event during which the ICS has ample capacity to remove the decay heat and depressurize the RPV and maintain fuel cooling for at least 72 hours. Fuel integrity is not a concern for large breaks. In the long-term Passive Containment Cooling System (PCCS) reduces containment pressure.

### **Small Breaks**

The smaller pipes connected to the RPV have an inside diameter  $\leq 19$  mm, do not have automatic RIVs, and are considered unisolated breaks. Small unisolated breaks may also occur in larger pipes. The leakage detection system detects breaks in large pipes connected to the RPV but may not be capable of detecting breaks that are smaller than the area of a circle with a 19 mm diameter. If the break is not detected, it is assumed to remain unisolated for 72 hours for CN sequences because no operator action is credited for this duration.

Small breaks conservatively credit only two of the three isolation condensers even though a break of less than 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser heat removal rate. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Not having sufficient steam in the RPV to feed the isolation condenser can only occur if the RPV is depressurized so far that almost all of the steam is escaping the break. This would be the case if the RPV pressure is even lower than that calculated for an instrument pipe break. However, in this case the break flow is less than the break flow calculated for an instrument pipe break, making the isolation condenser small break less limiting than the instrument pipe break. It can be concluded that whether the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser, the breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break.

For unisolated breaks, the RPV inventory is depleted faster as the break area becomes larger. The largest of the unisolated breaks is most limiting with respect to RPV inventory. Peak containment pressure is also higher as the break size is increased for an unisolated break.

### **Large and Small Pipe Break Summary**

The largest break sizes are the most limiting for isolatable (i.e., in the large break category) and un-isolatable (i.e., in the small break category) breaks.

In selecting the scenarios to evaluate for the pipe breaks, each of the sequences in the fault list is assessed with respect to the largest mass and energy release to containment and RPV inventory. Additional conservative assumptions that are described in Subsections 15.5.4.6 and 15.5.9 were made in constructing the bounding scenarios to reduce the number of analysis cases. Therefore, the bounding scenarios analyzed for pipe breaks in large and small break categories are scenarios bounding all scenarios in that category. However, the liquid and steam pipe breaks are analyzed separately.

The following common features are used in selecting the bounding sequences for pipe breaks:

1. A pipe break does not cause loss of feedwater or loss of normal containment cooling unless a direct or indirect effect of the pipe break causes the pump(s) to trip. Feedwater and normal containment cooling are lost concurrent with the break for the sequences involving a pipe break concurrent with Loss-of-Preferred Power (LOPP). This observation is used to determine whether the breaks are more limiting with or without LOPP.
2. All scrams credited in the LOCA analyses are direct scrams. A scram signal is initiated when the setpoint of the first scram function is reached for the scram functions that are available in the credited DL. Scram functions and setpoints in each DL and the trip parameters are provided in Subsections 15.5.4.6 and 15.5.9 and meet the guidance of CNSC REGDOC-2.4.1, Section 4.4.4.4.
3. In the CN-DBA sequences, a CCF of DL2 functions concurrent with the pipe break are assumed and credit only DL3 functions.

4. In the EX-DEC sequences, CCF of DL3 functions are assumed in the analysis and credit only DL2 and DL4a mitigation functions.

The guidelines for considering CCFs in the design and safety analyses are discussed in Chapter 3, Subsection 3.1.7.8.

The system responses, including the direct and indirect effects of large and small pipe breaks, are described in Subsections 15.5.4.6 and 15.5.9.

#### **15.2.4.6.1 Large Steam Pipe Breaks**

Large steam pipe breaks are postulated to occur in the following systems:

- Main Steam (MS) pipes
- ICS steam supply pipes

For breaks inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on detection of a steam pipe break (DL3-09).

MSRIV closes on high containment pressure for breaks inside containment (DL3-22), and on MS pipe break detection for breaks outside containment (DL3-20). ICS RIVs close for an ICS train when an ICS break inside or outside containment is detected in the respective ICS train (DL3-27, 28 or 29).

Because the isolation valves close rapidly, the effect on isolation condenser availability is not significant for large breaks mass and energy releases. After break isolation, one ICS is sufficient to remove decay heat and depressurize the RPV. As a bounding assumption, only one ICS train is credited for MS and ICS pipe breaks. This assumption is made so that an ICS steam supply break is bounded by an analyzed MS pipe break. One ICS train initiates on high containment pressure (DL3-15) for breaks inside containment. One ICS train initiates on pipe break indication detection (DL3-16) in MS or ICS pipes outside containment.

For a MS pipe break, the total break flow is the sum of break flows from both ends of the break. To bound all break locations, the break location is assumed as close to the RPV as possible, right outside the second or outboard RIV. Because two MS lines are connected through a header, the intact steam pipe also supplies the break location from the turbine side of the break. Break flow from the turbine side of the break is contributed by the flow from the RPV into the intact loop and the initial inventory in the piping. To maximize flow from the turbine side of the break, the isolation valves outside the containment are assumed to remain open, and TSV/TCV close rapidly. Because closure of CIVs outside containment are not credited, the calculated mass and energy release is applicable to breaks inside and outside containment.

If the break occurs when the plant is at very low power or hot shutdown, break flow from an MS pipe break may be higher due to carryover. At low power, there is more saturated water in the RPV to flash. This may increase the two-phase downcomer level much higher than that in the rated initial conditions case, contributing to the break flow due to increased liquid content although the break flow enthalpy is lower. In calculating the radiological consequences for breaks outside containment, the break mass flow rate for hot shutdown initial conditions may be more limiting. Both the rated initial conditions and the hot shutdown initial conditions are included in the bounding scenarios for MS pipe breaks.

The above scenario assumes LOPP concurrent with the break is bounding for the scenario when preferred power is available. If the preferred power is available, FW will continue to be injected to the RPV, which may increase mass release to the containment due to carryover. However, if preferred power is available, TSV/TCV also remains open, discharging much of the steam in the

intact loop to the turbine rather than to the break from the turbine side of the break location. As a result, the break scenario with LOPP is the more bounding scenario.

### **Bounding CN Scenario for Large Steam Pipe Breaks**

The bounding scenario analyzed MS pipe breaks inside and outside containment concurrent with LOPP:

The case is analyzed for rated initial power and hot shutdown conditions. The limiting break scenario is analyzed in Subsection 15.5.4.6.1.

#### **15.2.4.6.2 Large Liquid Pipe Breaks**

Large liquid breaks may occur in:

- FW pipe
- ICS condensate return pipe
- CUW pipe

The largest pipe from this list is the FW pipe and is analyzed in Subsection 15.5.4.6.2.

A large break in the ICS condensate return pipe is listed as a liquid break because the condensate return pipe is filled with water initially. However, this water is highly subcooled. The ICS condensate return valves are normally closed and the steam supply pipe is in communication with the RPV during normal operation. If a pipe break occurs at the condensate return pipe, the highly subcooled water in the condensate pipe is purged. The energy release to containment for a break inside containment or to the ICS pool for a break outside containment from the purged highly subcooled water is insignificant. After the liquid inventory in the pipe is depleted, the break flow becomes steam flow supplied from the RPV through the steam supply pipe until the RIVs close for the broken ICS unit.

The ICS steam supply pipe is fitted with a 70 mm inside diameter orifice at the steam distribution pipes. Therefore, flow from a break in the condensate pipe is only steam flow through a 70 mm orifice after the subcooled water is purged. This is less limiting than a break in the ICS steam supply pipe included in the large steam pipe break cases. This pipe break requires no further analysis because it is included in the large steam pipe break category for CN sequences.

CUW pipe breaks may occur inside or outside the containment. CUW pipe is a smaller-bore pipe and CUW breaks are isolated the same as the FW pipe breaks. Therefore, the CUW pipe breaks inside containment are bounded by the FW pipe breaks. CUW pipe breaks are routed through the same compartments that house the FW pipes in the reactor building. Therefore, CUW pipe breaks outside containment are also bounded by the FW pipe breaks for reactor building pressure and temperatures.

The limiting liquid pipe breaks inside and outside the containment are the FW pipe breaks. The break isolation, scram, and ICS initiating are discussed below.

### **Bounding CN Scenario for Large Liquid Pipe Breaks**

For a pipe break inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on pipe break indication in FW or ICS pipes (DL3-09). Scram does not occur for the CUW pipe breaks outside containment.

In accordance with the fault list, preferred power is available for CUW breaks outside containment and FW injection continues.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

LOPP concurrent with a break is assumed for the FW pipe breaks inside the containment. The TCV is conservatively assumed to close rapidly with LOPP. In the LOPP cases, there is flow from the pump side of the break before the CIVs close outside containment at 10 seconds due to pump coast down, but more importantly, due to flashing of the water in the FW piping. If the FW pumps continue running in spite of the break when preferred power is available, there is some increase in flow from the pump side of the break. However, normal containment cooling also continues to run when preferred power is available, compensating for the effect on containment pressure from an increase in the flow from the pump side of the break.

For breaks outside containment in the RB pressure and temperature calculations, the cases with and without FW running are included.

The radiological consequences for breaks outside containment uses the LOPP case that is more limiting. FW pump flow has no consequence because flow coming from the FW pump is decontaminated water and can only retard the break flow coming from the RPV through the intact loop. Only the water leaving the RPV is important in the radiological analyses that is higher if the pump is not running.

For breaks inside containment, RIVs close on high containment pressure (DL3-22). CIVs outside containment also close on high containment pressure (DL3-22).

For FW pipe breaks outside containment, FW RIVs close on break detection. MS RIVs also close on break detection (DL3-21).

For CUW pipe breaks outside containment, only CUW RIVs close on break detection (DL3-26).

For breaks inside containment, ICS is initiated on high containment pressure (DL3-15).

For FW and ICS pipe breaks outside containment, ICS is initiated on-line break detection (DL3-16). For CUW breaks outside containment, ICS initiation is not needed since preferred power is available.

### **Bounding Scenario Summary for Large Liquid Pipe Breaks**

The bounding liquid pipe break for containment response is analyzed for a FW pipe break inside containment:

- Double-ended guillotine FW pipe break inside containment concurrent with LOPP
- TSV/TCV are conservatively assumed to close rapidly retaining more energy
- FW pump trips and coast down
- Scram initiation within 1 second after the break
- ICs initiate on high containment pressure
- MS and FW RIVs start closing in 5 seconds and are fully closed in 10 seconds
- FW CIVs outside containment start closing in 5 seconds and are fully closed in 10 seconds
- FW conservatively assumed to trip at time zero and coasts down with a 3 second time constant

The bounding liquid pipe break scenario outside containment is similar to breaks inside containment with the following differences:

- CIVs are conservatively assumed not to close
- FW pump trip may or may not occur. Area pressure and temperature calculations consider both cases. The radiological analyses conservatively assume a FW pump trip

The above scenario does not rely on any scram or isolation function that is a result of the LOPP. Therefore, the same scenario bounds both preferred power available and LOPP cases.

#### **15.2.4.6.3 Small Breaks**

Small, unisolated steam or liquid pipe breaks may occur in instrument lines. Small breaks in the large pipes may also remain unisolated if they are below the leak detection system threshold, i.e., less than 19 mm inside diameter. The lowest location for a liquid pipe break is four (4) meters above the TAF.

Small breaks are analyzed using conservative assumptions demonstrating that fuel and containment integrity are maintained for at least 72 hours using only passive systems after which injection is recovered and the event is terminated. The LOCA acceptance criterion for demonstrating fuel integrity is to show that fuel cladding does not heat-up beyond normal operating temperature. This satisfies the fuel integrity acceptance criteria in Table 15.3-2 with large margin.

The fault list includes unisolated small breaks inside and outside the containment, with and without concurrent LOPP.

The bounding CN sequences in the fault list for a small break concurrent with LOPP are evaluated with respect to the fuel cladding and containment. For a break inside containment concurrent with LOPP, normal containment cooling system is also assumed to be lost. The energy discharged from the break to containment is removed by the PCCS (discussed in Chapter 6, Section 6.3.3) and through the containment dome. PCCS does not require actuation; it is always in service.

When the preferred power is available, FW continues to run. Because the FW pump can make up for the break flow, fuel integrity is not a concern. If preferred power is available, normal containment cooling also continues to run. If the normal containment cooling and PCCS cannot keep up with the break flow and containment pressure increases, the reactor scrams, isolation condensers initiate and RPV depressurizes reducing the break flow. Containment cooling continues to operate maintaining containment pressure at a lower value than PCCS alone maintains in the LOPP case.

The bounding small liquid and steam pipe break scenarios are the same, the only difference is the discharge from the small liquid pipe break is initially from the liquid water space. It becomes steam flow after level falls below the RPV nozzle elevation of the broken pipe.

Following an unisolated break in an instrument pipe concurrent with LOPP, turbine pressure decreases rapidly resulting in a decrease in the steam pipe pressure and an increase in steam flow. Although a consequential closure of TCVs may occur on LOPP, this is not credited in the analysis. TCV is assumed to remain in its initial position. Scram is initiated when the steam pipe pressure decreases to low steam pipe pressure setpoint (DL3-02) with a 1.7 second delay. Power is assumed to remain at the initial value for an additional 2 seconds to account for the time elapsed until prompt fission power is diminished after the control rods start inserting. MSRIVs also start closing on low steam pipe pressure (DL3-17) over 5 seconds and are fully closed in 10 seconds.

FW pumps trip concurrent with LOPP and coast down with a 3 second time constant.

All available ICS train condensate return valves start opening with a delay of 1 second when the level falls to L2 level setpoint (DL3-14). Only two isolation condenser trains are assumed available. ICS condensate return valves are fully open 10 seconds after they start opening. There are no further actuations assumed for the remainder of the event.

Mass and energy releases from small pipe breaks do not credit containment back pressure for breaks inside containment. In addition, reactor scram and isolations initiated by high containment pressure are also not credited. Therefore, the above scenario applies to breaks outside containment as well as breaks inside containment. Because a break outside containment occurs from a longer pipe, break mass and energy releases calculated for a break inside containment bounds a small break outside containment.

### **Bounding Small Pipe Break**

The bounding scenario is summarized as follows:

- Small steam or liquid pipe break concurrent with LOPP
- FW pump trips and feedwater flow coasts down with a time constant of 3 seconds
- Pressure controller failure, TSV/TCV position remain open at their initial position
- Reactor scrams when steam pipe pressure decreases to low steam pipe pressure setpoint with a 1.7 second delay
- MSRIV closure when steam pipe pressure decreases to low steam pipe pressure setpoint with a 5-second delay
- ICS initiation when level is less than L2

### **Bounding Scenarios for DEC Pipe Breaks**

DECs assume DL3 CCF in addition to the pipe break. Only DL2 and DL4a functions are credited in DECs. There is either a DL2 or DL4a function for all credited DL3 functions in Table 15.5-49, except the isolation condenser pipe breaks. Because the heat removal by the ICs is a higher-class safety function than isolation by a DL4a function, no DL4a associated function exists. An unisolated isolation condenser pipe break is similar to an unisolated MS pipe break and is evaluated separately subject to different acceptance criteria.

DL2 and DL4a functions performing the same function for the credited DL3 functions are listed in Table 15.5-50. The setpoints and timing of these functions are the same as or close to the DL3 functions. Therefore, except for the isolation condenser pipe breaks, the analyzed design basis LOCA analyses bound the DEC pipe breaks for the isolatable large pipe breaks and un-isolatable small breaks.

#### **15.2.5 References**

- 15.2-1 IAEA Safety Standards Series No. SSR-1, "Site Evaluation for Nuclear Installations Safety Requirements," International Atomic Energy Agency.

**Table 15.2-1: Fault or Event Groups and Explanation of Core Reactivity Response Basis**

Name (ID)	Description	Discussion of Reactivity Effects
Temperature Decrease (TD)	Decrease in Core Coolant Temperature	Void reactivity is key. The decrease in temperature results in an increase in the core inlet subcooling. More core thermal power goes into heating up the water and less into void production. The core void fraction decreases and causes the power to increase. This is a relatively slow increase in core power due to the thermal inertia of the coolant.
Pressure Increase (PI)	Increase in Reactor Pressure	Void reactivity is key. Pressurization results in decrease of core voids and increase in core power. For rapid pressurization, control rod scram is needed to mitigate. Core exposure effects on void reactivity and control rod position are important aspects that are modeled and included in the analysis.
Reactivity Increase (RI)	Reactivity and Power Distribution Anomalies	Control rod reactivity is key. Errors or failure in control rod movement are expected event initiators as these events add reactivity and change the local and core wide power. Fuel loading errors are also included.
Inventory Increase (II)	Increase in Reactor Coolant Inventory	Void reactivity is key. An increase in coolant inventory results in a reactor water level increase. The increase in reactor water level normally has the following effects: <ul style="list-style-type: none"> <li>- Core flow increase (small reactivity effect at rated power conditions)</li> <li>- Core inlet subcooling increase (because the additional inventory is expected to be from lower temperature coolant, larger reactivity effect than the core flow increase)</li> </ul> These effects tend to increase void reactivity.
Inventory Reduction (IR)	Decrease in Reactor Coolant Inventory	Void reactivity is key. The decrease in reactor water level has the following potential results: <ul style="list-style-type: none"> <li>- Core flow decrease (small reactivity effect at rated power conditions)</li> <li>- Core pressure decrease (if break in coolant pressure boundary cannot be compensated for by pressure control)</li> <li>- Core inlet subcooling decrease (comes with pressure decrease)</li> </ul> These effects tend to insert negative reactivity, resulting in core power decrease, due to void reactivity. It is typical in loss-of-coolant accident analysis to ignore these effects as the protection systems typically act quickly to insert control rods and shutdown the core before the negative void reactivity feedback has significant effect.
Non-Reactor Faults	Event specific. These events are not core related.	Fuel Handling Accident



**Table 15.2-2: Bounding Events Transient (Non-LOCA) and LOCA**

DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
<b>Decrease in Core Coolant Temperature Bounding Event Summary</b>		
BL-AOO	<b>Loss of Feedwater Heating (LFWH)</b> TD-LFWH_BL-AOO	15.5.3.1.1
CN-DBA	<b>Common Cause Failure – Loss of Feedwater Heater</b> CCF- LFWH, Passive CCF DL2 Technology Platform; TD-CCF- LFWH_CCF-DL2_CN-DBA	15.5.4.1.1
EX-DEC	None	N/A
<b>Increase in Reactor Pressure Bounding Event Summary</b>		
BL-AOO	<b>Generator Load Rejection or Turbine Trip</b> (LR-TT); PI-LR-TT_BL-AOO	15.5.3.2.1
	<b>Closure of One MSRIV</b> 1MSRIVC; PI-1MSRIVC_BL-AOO	15.5.3.2.2
	<b>Loss of Condenser Vacuum (LOCV)</b> PI-LOCV_BL-AOO	15.5.3.2.3
	<b>Loss-of-Preferred Power (LOPP)</b> PI-LOPP_BL-AOO	15.5.3.2.4
CN-DBA	<b>Load Rejection or Turbine Trip</b> LR-TT, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LR-TT_CCF- DL2_CN-DBA	15.5.4.2.1
	<b>Loss-of-Preferred Power</b> LOPP, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LOPP_CCF- DL2_CN-DBA	15.5.4.2.2
	<b>RPV Pressure Control Downscale</b> CCF - RPV Pressure Control Downscale (CCF-RPCD), Passive CCF DL2 Technology Platform (CCF-DL2); PI-CCF- RPCD_CCF-DL2_CN-DBA	15.5.4.2.3
	<b>Closure of All MSRIVs and FW Isolation</b> CCF - Closure of All MSRIVs and FW isolation valves (CCF-DL4a-MSRIVC- FWIV); PI-CCF-DL4a-MSRIVC- FWIV_CN-DBA	15.5.4.2.4

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
EX-DEC	<b>Closure of One MSRIV</b> PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC <b>Complex Sequence of Generator Load Rejection or Turbine Trip</b> Complex Sequence of LR-TT + CCF-Mechanical-Scram; CSS-LR-TT_CCF-Mechanical-Scram_EX-DEC <b>Loss of Condenser Vacuum</b> LOCV, CCF-Hydraulic-Scram; PI-LOCV_CCF-Hydraulic-Scram_EX-DEC <b>Loss-of-Preferred Power</b> LOPP, CCF-Hydraulic-Scram; PI-LOPP_CCF-Hydraulic-Scram_EX-DEC	15.5.5.2.1   15.5.5.2.2   15.5.5.2.3   15.5.5.2.4
<b>Reactivity and Power Distribution Anomalies Bounding Event Summary</b>		
BL-AOO	None	N/A
CN-DBA	<b>Fuel Loading Error (FLE)</b> RI-FLE_CN-DBA	15.5.4.3.1
EX-DEC	<b>CCF- All Control Rod Withdrawal at Power - All Rods (CCF-ACRW)</b> Passive CCF DL2 Technology Platform (CCF-DL2); RI-CCF-ACRW_CCF-DL2_EX-DEC <b>Inadvertent Control Rod Withdrawal at Power - Single rod (ICRW)</b> RI-ICRW_DL2-CCF_EX-DEC	15.5.5.6.1  15.5.5.6.2
<b>Increase in Reactor Coolant Inventory Bounding Event Summary</b>		
BL-AOO	<b>Inadvertent Isolation Condenser Initiation – One Train (IICI-1)</b> II-IICI-1_BL-AOO	15.5.3.4.1

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
CN-DBA	<b>Feedwater Flow Increase – All Pumps</b> CCF-FWFI with Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF_FWFI_CCF-DL2_CN- DBA	15.5.4.4.1
	<b>Inadvertent Isolation Condenser Initiation - All Trains</b> (CCF-DL4a-II-ICI), Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF-II-ICI_CCF-DL2_CN- DBA	15.5.4.4.2
EX-DEC	None	N/A
<b>Decrease in Reactor Coolant Inventory Bounding Event Summary (non-LOCA)</b>		
BL-AOO	<b>Feedwater Pump Trip – One Pump</b> FWPT; IR-FWPT_BL-AOO	15.5.3.3.1
CN-DBA	<b>CCF Loss of FW Flow</b> Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-LOFW_CCF- DL2_CN-DBA	15.5.4.5.1
	<b>Reactor Pressure Vessel Pressure Controller Open</b> CCF-RPCO, Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF- RPCO_CCF-DL2_CN-DBA	15.5.4.5.2
EX-DEC	<b>Feedwater Isolation</b> FW Isolation (CCF-FWI-DL3); IR-CCF- FWDI-DL3_EX-DEC	15.5.5.8
<b>Decrease in Reactor Coolant Inventory Bounding Event Summary (LOCA)</b>		
CN-DBA	<b>Main Steam Pipe Breaks Inside the Containment, Conservative Case</b>	15.5.4.6.1
CN-DBA	<b>Feedwater Pipe Break Inside the Containment, Conservative Case</b>	15.5.4.6.2
CN-DBA	<b>Large Isolation Condenser Pipe Breaks Inside the Containment</b>	15.5.4.6.3
CN-DBA	<b>Small Steam and Liquid Pipe Breaks Inside the Containment</b>	15.5.4.6.4
CN-DBA	<b>Large Main Steam Pipe Break Outside the Containment</b>	15.5.9.2.1
CN-DBA	<b>Large Feedwater Pipe Break Outside the Containment</b>	15.5.9.2.2

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>DSA Layer / Event Category</b>	<b>Event and Fault Sequence ID</b>	<b>Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results</b>
CN-DBA	<b>Large Isolation Condenser Pipe Breaks Outside the Containment</b>	15.5.9.2.3
CN-DBA	<b>Small Breaks Outside the Containment</b>	15.5.9.2.4

### **15.3 Safety Objectives and Acceptance Criteria**

Implementation of the safety objectives established by the IAEA Safety Standards Series No SF-1 Fundamental Safety Principles ensure that the BWRX-300 facility when operated achieves the highest standard of reactor safety that can be reasonably achieved. The IAEA SF-1 Safety Objectives are discussed in Chapter 3, Subsections 3.1.1. and 15.1.2.

The deterministic and probabilistic safety analysis acceptance criteria are based on or derived from ensuring that the dose acceptance criteria of CNSC REGDOC-2.4.1, Section 4.3.2 (Reference 15.3-1), and CNSC REGDOC-2.5.2, Section 4.2.1 (Reference 15.3-2) are met.

#### **15.3.1 Deterministic Safety Analysis Acceptance Criteria**

The BWRX-300 design complies with acceptance criteria of CNSC REGDOC-2.4.1, Section 4.3.2 (Reference 15.3-1), and the established dose acceptance criteria in CNSC REGDOC-2.5.2 Section 4.2.1.

CNSC REGDOC-2.5.2 (Reference 15.3-2) states “that acceptance criteria shall be assigned to each plant state in the design, considering the principle that frequent PIEs have only minor or no radiological consequences, and that any events that may result in severe consequences are of extremely low probability.”

Qualitative acceptance criteria are defined and met for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. Safety goals are defined in REGDOC-2.5.2, Section 4.2.2 and are discussed in Chapter 3, Section 3.1.2.

Derived qualitative and quantitative acceptance criteria are used to analyze AOOs or DBAs. Qualitative acceptance criteria are supported by experimental data, prescribed by regulatory requirements, or prescribed by applicable codes and standards. The results of the quantitative safety analysis confirm the derived acceptance criteria (i.e., the limiting event in an event group).

Certain accidents with a predicted frequency of occurrence less than  $1\text{E-}5/\text{rx-yr}$  DECAs may be used as design basis events for a safety system. In this case, the results are compared to the DBA dose limits, and qualitative acceptance are established. The DBA derived acceptance criteria are used as screening criteria to evaluate core damage and are used as input in determining the PSA safety goals.

The committed whole-body dose for average members of the critical groups who are most at risk, is calculated in the DSA for a period of 30 days after the analyzed event. As stated in Section 3.1.2, the calculated dose is less than or equal to the dose acceptance criteria of CNSC REGDOC-2.5.2, Section 4.2.1:

- 0.5 millisievert (mSv) for any AOO
- 20 mSv for any DBA

The dose results provided in Section 15.7 from the limiting events identified in Section 15.5 demonstrate that the radiological consequences of the analyzed events do not exceed the AOO, and DBA acceptance criteria listed in Tables 15.3-1 and 15.3-2, respectively.

##### **15.3.1.1 Acceptance Criteria for Analysis of Anticipated Operational Occurrences**

CNSC REGDOC-2.4.1, Section 4.3.2 requires derived acceptance criteria be established for AOOs and DBAs per CNSC REGDOC-2.4.1, Section 4.3.4. The derived acceptance criteria for the DSA of AOOs are shown in Table 15.3-1. These derived acceptance criteria are based upon the fission product barrier or FSF.

#### **15.3.1.2 Acceptance Criteria for Analysis of Design Basis Accidents**

CNSC REGDOC-2.4.1, Section 4.3.2 requires derived acceptance criteria be established for AOs and DBAs per CNSC REGDOC-2.4.1, Section 4.3.4. The derived acceptance criteria for the deterministic safety assessment of DBAs are shown in Table 15.3-2.

#### **15.3.2 Acceptance Criteria for Probabilistic Safety Assessment**

The safety goals for the PSA of core damage frequency, small release and large release frequencies are shown in Table 15.3-3 and are consistent with the quantitative safety goals in CNSC REGDOC-2.5.2, Section 4.2.2.

#### **15.3.3 References**

- 15.3-1 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 15.3-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 15.3-3 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 15.3-4 NEDC-33840P, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," GE-Hitachi Nuclear Energy Americas, LLC.

**Table 15.3-1: Anticipated Operational Occurrence Deterministic Safety Assessment Acceptance Criteria**

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	An AOO will not escalate to a more serious plant condition unless other faults occur independently.	Not applicable
	There is no loss of function of any fission product barrier.	Not applicable
Fuel Rod	Loss of fuel rod mechanical integrity will not occur due to fuel melting.	The calculated maximum fuel center temperature $T_{\text{center}}$ remains below the fuel melting point $T_{\text{melt}}$ . Subsection 4.2.3.4 describes the method used in calculating the maximum fuel pellet temperature.
	Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction.	The cladding strain acceptance criteria defined in Section 5.0 of Reference 15.3-4. Chapter 4, Subsection 4.2.3.4 describes the code methodology used in calculating the cladding strain acceptance criteria.
	Fuel rod failure will not occur due to overheating of cladding	The calculated core Minimum Critical Power Ratio (MCPR) ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition during AOO events. Chapter 4, Section 4.4.1 describes the establishment of the fuel cladding safety limit. With the reactor steam dome pressure less than 4.72 MPaG (685 psig), the calculated reactor thermal power is less than 25% of rated thermal power.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient.	The calculated peak pressure associated with the reactor coolant pressure boundary shall not exceed 110% of the design pressure or 11.38 MPaG (1650 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	The calculated reactor water level is maintained at or above TAF.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment integrity is maintained. If an AOO results in an energy release to the containment, or loss of containment heat removal, then containment stresses (i.e., pressure and temperature) are limited such that there is no loss of a containment barrier safety function, and thus, the containment remains within its design limit values.	No AOOs result in a significant energy release to containment, or prolonged loss of normal containment cooling. The normal operation limits and conditions are applied to containment, and no AOO containment quantitative criteria is needed. Chapter 9, Subsection 9A.5.6 describes the containment cooling system functional design.
Long-Term Heat Removal	SSC important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all Operational States, including AOOs.	Following AOO events that do not result in shutdown, a controlled condition is achieved.  Following AOO events that require shutdown, the core remains shutdown independent of operator action or offsite support for at least 72 hours.  AOO events that rely on DL3 mitigation for long-term cooling are capable of providing cooling for at least 72 hours without operator action or offsite support.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.3-2: Design Basis Accident Acceptance Criteria**

<b>Fission Product Barrier or Fundamental Safety Function</b>	<b>Qualitative Acceptance Criteria</b>	<b>Quantitative Acceptance Criteria</b>
General	Except for fuel cladding, there is no loss of function of any fission product barrier.	Not Applicable
Fuel Rod Failure	The number of fuel rod failures is conservatively estimated for DBAs.	The calculated number of failed rods does not result in exceeding the applicable radiological dose acceptance criteria.
	Mechanical fracturing of a fuel assembly under DBA loading conditions does not result in losing the ability to cool the fuel assembly.	The mechanical integrity of the fuel is established from the mechanical and thermal fuel analysis described in Chapter 4, Subsection 4.2.2.
Fuel Cooling	The calculated fuel cladding temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.	The calculated PCT remains less than 1204°C (2200°F). The calculated total oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation for DBAs where exceeding the oxidation thickness challenges the capability to cool the core.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient as a result of a DBA.	The calculated peak pressure associated with the RCPB shall not exceed 120% of the design pressure or 12.41 MPaG (1800 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	Conformance is demonstrated by meeting the fuel cooling and long-term heat removal criteria.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment pressures and temperatures are maintained below the design values.	The calculated containment pressure does not exceed the design pressure 0.414 MPaG (60 psig). The calculated containment shell temperature does not exceed the design temperature 165.6°C (330°F).
	The local combustible gas concentrations in the containment are within the range where deflagration or detonation cannot occur.	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
	Containment energy management systems are capable of reducing the containment pressure and temperature following a DBA to minimize the release of fission products to the environment and to preserve containment integrity and leak tightness.	The calculated containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting LOCA within 24 hours.
Reactivity Control	Reactivity control required to bring the reactor to cold shutdown is maintained.	Shutdown margin is established to assure that the reactor can be brought subcritical with the highest-worth control rod pair withdrawn when the core is in its most reactive condition. The subcriticality value is 0.38% $\Delta k/k$ with the highest-worth control rod pair analytically determined.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all operational states, and DBAs.	Long-term cooling is maintained for a minimum of 72 hours independent of operator action and offsite support, and for 30 days with credit for operator actions and on-site resources.  For DBA events that result in shutdown, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 215.6°C (420°F).

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.3-3: Probabilistic Safety Goals**

<b>Qualitative Acceptance Criteria</b>	<b>Derived Quantitative Acceptance Criteria</b>
Core damage frequency	The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than $1\text{E-}5/\text{rx-yr}$ .
Small release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than $1\text{E}15$ becquerels of iodine-131, shall be less than $1\text{E-}5/\text{rx-yr}$ .
Large release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the local population or a release to the environment of more than $1\text{E}14$ becquerels of Cesium-137 shall be less than $1\text{E-}6/\text{rx-yr}$ .

## **15.4 Human Actions**

The BWRX-300 design approach minimizes the complexity of the SSC design, while enhancing reliability and reducing the potential for human error. The systems and components are designed to reduce the necessity for human actions (refer to Chapter 18 Human Factors Engineering). Where complexity is necessary in the design (e.g., self-diagnostic tools, redundancy in equipment in a single division), the design complexity is documented and justified for enhancing reliability, surveillance, calibration, and other equipment attributes.

Chapter 18 discusses the human factors engineering and human-machine interface consideration during development of the facility design that facilitate interaction between operating personnel and the plant. Subsection 15.1.3 discusses the human operation hazard evaluation under the safety assessment framework.

Chapter 13 provides a program description to manage operational aspects that are affected by human factor considerations, including the continued review and development of measures in place. The chapter also describes the organizational provisions that ensure operators are able to effectively perform in the main and secondary control rooms and other parts of the plant under all operational circumstances, including proposed shift schemes and rotations, assessment of operator's fitness for duty, and other human factors related issues.

### **15.4.1 Human Actions in Deterministic Safety Analysis**

There are no operator actions credited in responding to the events analyzed in Section 15.5 for the DSA. This assurance is attributed to the following design features:

1. Fail-safe (not reliant on external power) safety system actuations ensure that the FSFs are fulfilled for a DBA
2. Automatic, reliable actuation of the control rods with either stored energy or motors to shut down the reactor and maintain it in a guaranteed shutdown state via latching mechanisms
3. ICS provides passive decay heat removal
4. Fail-safe containment isolation and passive containment heat removal

Critical safety parameter monitoring is provided in the Secondary Control Room (SCR) if the Main Control Room (MCR) becomes uninhabitable.

The HOHE identifies failures that involve an erroneous decision or action taken by a human that can lead to an unplanned plant transient that is evaluated as a PIE in the hazard analysis in Section 15.1.3.

### **15.4.2 Human Actions in Probabilistic Safety Assessment**

Human actions resulting from PSA event evaluations are discussed in Subsection 15.6.1.3.5.

## **15.5 Deterministic Safety Analysis**

### **15.5.1 General Description of the Approach**

The DSA is divided into two parts:

- Part One – the plant response to fault sequences is evaluated and analyzed to confirm the performance of the fission product barriers against the derived acceptance criteria
- Part Two - the event dose consequences resulting from a fission product release or other source of radiation, such as the reactor coolant, is radiologically analyzed

The DSA is performed based on the outputs of the hazard analysis and fault evaluations. There are three layers of DSA performed (see 15.1 and 15.2.1 for additional details) that credit different sets of DLs:

#### *Deterministic Safety Analysis Approach for Non-LOCA Events*

Transient DSA analyzes fault sequences where the reactor coolant pressure boundary remains intact. These events are broken down into groups that result in similar core responses. Section 15.2.4 describes the core response during off-normal conditions, the groups determined for BWRX-300 (Table 15.2-1), and the selected bounding event scenarios for AOO, DBA and DEC without core damage analyses (summarized in Table 15.2-2).

The BWRX-300 scenarios are identified through the fault evaluations. The methods and assumptions described in the TRACG Application for BWRX-300 (Reference 15.5-3) are used to confirm the performance of the fission product barriers for the DSA non-LOCA events. The TRACG Application for both non-LOCA and LOCA event analysis is discussed in Subsection 15.5.1.2.1.

The TRACG Application for BWRX-300 also includes the stability analysis that evaluates potential coupled thermal-hydraulic – neutronic instabilities in the reactor core. TRACG is used to perform transient safety analysis and stability analysis for both forced flow and natural circulation BWR designs. Previous TRACG applications as well as BWRX-300 use the systematic approach developed for the US NRC, called Code Scaling, Applicability and Uncertainty (CSAU) to confirm the applicability of a computer code for DSA. This approach involves systematic evaluation of the phenomena that are important for the plant design and accident scenarios identified. A qualitative process is used to identify and rank the importance of phenomena. Through this process a Phenomenon Identification and Ranking Table (PIRT) is established. The PIRT is used together with the TRACG documentation to systematically demonstrate the applicability of TRACG models and the qualification of the TRACG model to predict the phenomena. Defining the nodalization and evaluation of the effects of scale are included. In addition to code applicability and qualification, the PIRT is also used as the basis to perform quantitative uncertainty analysis of transient scenarios, if needed. Additional information regarding the approach for addressing uncertainty in the DSA is provided in Subsection 15.5.1.1

The TRACG applicability to model phenomena also requires that the code capability be demonstrated to apply the code in the intended manner with a qualifying result achieved. TRACG capability to model phenomena is important to BWRX-300 simulation and is consistent with modern best practices. TRACG qualification is based upon proven practices for verification and validation using acceptable codes and standards (see Chapter 3, Appendix 3G). Experiments and plant events used to validate TRACG provide evidence that TRACG can be applied for the BWRX-300 design.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Integral to the capability of TRACG for transient DSA is the use of three-dimensional nuclear kinetics input. This input comes directly from and essentially uses the same methods as the steady-state core simulator, PANAC11. PANAC11 is used in the BWRX-300 as described in Subsections 4.3.2 and 15.5.1.2.3. Other code interfaces are described in TRACG Application for the BWRX-300 (Reference 15.5-3).

Design control procedures require independent verification of safety analysis calculations to ensure that results are properly summarized from calculations, physically sound/correct, and consistent with expected results when compared to previous calculations. The results are then confirmed to meet the appropriate acceptance criteria. Table 15.5-48 "Conservatisms Used in the Non-LOCA DSA" provide additional insight in conservatisms used in the transient analysis.

*Deterministic Safety Analysis Approach for LOCA Events*

The methods and assumptions of the DSA confirming the performance of the fission product barriers for LOCAs are described in Licensing Topical Report (LTR) NEDC-33922P-A, Revision 3, BWRX-300 Containment Method (Reference 15.5-2).

TRACG calculates the mass and energy release from modeled breaks of various sizes and locations. Atmospheric pressure is used for the TRACG pressure boundary condition for any breaks. This approach provides no credit for the back pressure from containment. Consequently, the retained RPV inventory calculated by TRACG represents the minimum coolant volume. This modeling provides results as if the break occurred outside containment.

Breaks inside containment realistically experience back pressure from containment that reduces the mass and energy calculated by TRACG once the break flow becomes unchoked. However, this effect is not treated explicitly because it requires two-way coupling between the TRACG calculation and the GOTHIC containment calculation. Instead, the methodology has a one-way coupling with the mass and energy release rates conservatively calculated by TRACG that supplies inputs to the GOTHIC calculation up until the point in time when the containment and RPV pressures first equalize. Choked flow naturally satisfies the assumed one-way coupling because choked flow does not depend on the downstream pressure. Select TRACG inputs are specified so that mass and energy release rates are conservatively calculated.

Rapid mass and energy releases into containment occur before a large break is isolated. This leads to the highest containment peak pressure at approximately the same time that the break is isolated. For large breaks, the containment shell is the dominant short-term energy sink, and it causes containment pressure to decrease from its peak value after isolation of the break occurs.

Compared to large breaks, small unisolated breaks have a much slower mass and energy release rate from the RPV into containment. The lowest break on the RPV that remains unisolated and occurs outside containment produces the most limiting scenario for minimum RPV inventory. Regardless of break location and whether it is inside or outside containment, break flow slowly decreases with time because the RPV is being depressurized largely due to the ICS and to a much lesser extent by the break flow.

Containment pressure slowly increases and eventually equals the RPV pressure for breaks inside containment. It is not realistic to use the TRACG break flow that was calculated using an atmospheric pressure boundary condition as input to the GOTHIC containment calculation after the point in time when containment and RPV pressures first equal each other. A better approximation is to assume zero break flow after this point in time, but this could potentially be nonconservative with respect to the longer-term calculated containment pressures. The proposed methodology does not require the GOTHIC calculation to continue beyond the point where the containment and RPV pressures equalize, because the longer-term containment pressure is bounded by the RPV pressure calculated.

The methods and assumption for normal coolant radiological analyses are described for these events in Section 15.7.

#### **15.5.1.1 Safety Margins in Safety Analysis**

The safety margin is the result of the conservative assumptions used in the analysis and design rules applied to the SSC design capabilities. In addition, the DSA demonstrates that the challenges to the physical barriers do not exceed their physical capacity.

Uncertainties in initial conditions and methods are accounted for in the CN-DSA. The BL-DSA and EX-DSA allow best-estimate methods to be consistently applied using REGDOC-2.4.1, Section 4.4.2 guidance. For CN-DSA thermal-hydraulic analysis, a graded approach is used in combining uncertainties. The graded approach involves a qualitative assessment of the safety margin on a case-by-case basis and includes a review of the magnitude of results compared to acceptance criteria along with the judgment of conservatism in the derived acceptance criteria.

The DSA confirms the FSFs successfully keep plant radioactive material releases within the acceptance criteria with adequate safety margins.

##### **15.5.1.1.1 Large Margin**

For events with large margin or substantially non-limiting, there is no need to apply uncertainty to the analysis methodology. Judgment is used to establish what is “large margin” or “substantially non-limiting”. Instead of a quantitative evaluation of uncertainty, the event is dispositioned qualitatively based on the uncertainty evaluation performed for a limiting event of a similar type, historical analysis of similar type, or other qualitative based disposition.

#### **Large Margin Examples**

The inadvertent isolation condenser initiation in Subsection 15.5.4.4.2 and the closure of all MSRIVs and FWIVs in Subsection 15.5.4.2.4 are examples of large margin events. There are many DBA events that have minimal impacts compared to the acceptance criteria.

##### *Inadvertent Isolation Condenser Initiation*

This CN-DBA event results in a peak pressure of 7.32 MPaG, much lower than the acceptance criteria of 12.41 MPaG and is bounded by pressure increase CN-DBA events in Subsection 15.5.4.2.1. Also, the PCT is 307.9°C, much lower than the acceptance criteria in Table 15.3-2 and is bounded by the generator load rejection CN-DBA event in Subsection 15.5.4.2.1. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary.

##### *Closure of All Main Steam Reactor Pressure Vessel Isolation Valves and FW Isolation Valves*

This CN-DBA event results in a peak pressure of 8.73 MPa, much lower than the acceptance criteria of 12.41 MPaG. The peak pressure is bounded by other pressure increase CN-DBA events in Subsection 15.5.4.2. Also, the PCT is 312.47°C, much lower than the acceptance criteria in Table 15.3-2. This event has a large safety margin to the DBA acceptance criteria. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary. As a result, for the above events and any other events with similar margin, there is no need for any quantification of uncertainty and a qualitative disposition is adequate.

#### **15.5.1.1.2 Medium Margin**

In medium margin scenarios, method uncertainty is addressed by biasing key important phenomena in a conservative direction (typically one or two sigma). Input parameters such as power, pressure, level, temperature are based on using the most limiting normal operating values. In these cases, the selection of key important phenomena is dependent on the specific event evaluated. Important phenomena can be different for the output parameters when multiple output parameters are considered for selecting the bias direction. The selection of the key important phenomena and determining the bounding bias direction is considered for each output parameter that has medium margin and compared to the derived acceptance criteria.

#### **Medium Margin Example**

A medium margin event is the large break inside containment described in Subsection 15.5.4.6.

#### ***Large Pipe Breaks Inside Containment***

There are multiple large pipe breaks inside containment event scenarios evaluated that have commonalities. These events result in no significant fuel cladding heat-up and are not bounding with respect to maintaining inventory above the fuel (to ensure continued cooling). They represent the largest challenge to the containment fission product barrier. These events are treated as medium margin events and the initial conditions and modeling parameters are biased to ensure conservative containment conditions are calculated. The initial conditions used are provided in Table 15.5-1. The modeling parameters biased in this analysis are discussed in NEDC-33922P (Reference 15.5-2). The combination of the conservative biased inputs combined with the observation of the margin available results in conservative analyses.

#### **15.5.1.1.3 Low Margin or Quantitative Evaluation of Uncertainty is Desired**

A proven Monte Carlo technique is used to combine the individual biases and uncertainties into an overall bias and uncertainty for low margin events. This process is described in the TRACG Application for BWRX-300 (Reference 15.5-3). There are no events identified as low margin.

#### **15.5.1.2 Description of the Computer Codes Used in the Safety Analyses**

There is a large amount of data available from operating BWR plants and from the testing and licensing efforts to licence the predecessor BWR/Advanced Boiling Water Reactor (ABWR)/Economic Boiling Water Reactor (ESBWR) designs and individual plants. The vast database of feature performance in licensed reactors, combined with the recent thorough licensing review of the ABWR and ESBWR performed by the USNRC provides an extremely well qualified foundation from which to make the modest extrapolations to the BWRX-300. The following codes, methods, and accompanying assumptions are used in evaluating the performance of the BWRX-300 during postulated initiating events. The radiological responses to DBAs and DECAs are presented in Section 15.7 but use USNRC-accepted RADTRAD and PAVAN codes for doses at the exclusion boundary and atmospheric dispersions, respectively. A description of codes used in the BWRX-300 safety analysis is provided in Chapter 3, Appendix 3G.

#### **15.5.1.2.1 TRACG**

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG is the primary licensing analysis tool for LOCA and transient analyses for PIEs with a large range of frequencies up to events that do not involve significant core damage (severe accidents). TRACG has been used in a variety of applications for operating BWRs as well as design/analysis for the ESBWR.



TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the operation of BWRs. It is a best-estimate code for analysis of BWR transients ranging from simple operational transients to design basis LOCAs and failure to scram transients. TRACG has an extensive qualification base for separate effects, BWR fuel and components, and integral tests. It has been reviewed and approved by the US NRC for a number of analysis applications such as AOOs, ECCS/LOCA and failure to scram overpressure (a BDBA event for BWRs) analyses.

TRACG is used to analyze the challenges to the fuel, RPV, and the mass and energy releases to the containment, for LOCA and non-LOCA DSA. TRACG draws from the licensed BWRs database, which includes design features of the BWRX-300 (albeit in various configurations) and appropriate testing and allows direct application to BWRX-300 design and analysis. TRACG is maintained and updated by GEH.

### **Scope and Capabilities**

TRACG is based on a multi-dimensional two-fluid model for the reactor thermal-hydraulics and a three-dimensional neutron kinetics model. The two-fluid model used for the thermal-hydraulics solves the conservation equations for mass, momentum, and energy for the gas and liquid phases. TRACG does not include any assumptions of thermal or mechanical equilibrium between phases. The gas phase may consist of a mixture of steam and a non-condensable gas, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum, and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime-dependent and are determined based on a single flow regime map, which is used consistently throughout the code. In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for BWR components, such as fuel channels, steam separators, and can simulate BWR steam dryers as part of its vessel model. TRACG also contains a control system model capable of simulating the major BWR control systems such as those for pressure and water level.

The neutron kinetics model is consistent with the GEH BWR Core Simulator PANACEA. It solves a modified one-group diffusion model with six delayed neutron precursor groups. Feedback is provided from the thermal-hydraulic model to the kinetics model for moderator density, fuel temperature, boron concentration and control rod position.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, valve, tee, channel, steam separator, heat exchanger and vessel. System simulations are constructed using these components as building blocks. Any number of these components may be combined. The number of components, their interaction, and the detail in each component are specified through code input. Consequently, TRACG has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete BWR plants.

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and full-scale BWR plant data. A detailed documentation of the TRACG qualification is contained in the TRACG Qualification Report (Reference 15.5-1).

The total effort and extent of qualification performed on TRACG, since its inception in 1979, now exceeds, both in extent and breadth, that of any other engineering computer program GE/GEH has submitted to the USNRC for design application approval.

The analysis also determines the most limiting overpressure protection events in terms of peak vessel pressure. The results are used to demonstrate adequate pressure margin to the reactor vessel design limit.

### **Scope of Application of TRACG to BWRX-300**

The PIRT discussed in References 15.5-2 and 15.5-3 identify specific governing phenomena, where a significant fraction were concluded to be "important" in predicting BWRX-300 transient and LOCA performance. Most of these phenomena are common to operating BWRs. This section examines specific SBWR/ESBWR-related tests and test facilities beyond the previous qualification database. Early in the SBWR program, it was identified that there was no information in the data base for a heat transfer correlation for steam condensation in tubes in the presence of non-condensable gases. A Single Tube Condensation Test Program was conducted to secure this information and reported to the USNRC in TRACG Qualification Report for SBWR (NEDC-32725P, Revision 1) and ESBWR NEDC-33080P-A, Revision 2) that are used in the TRACG model for the BWRX-300 as described in LTR NEDC-33922P (Reference 15.5-2).

The test program was conducted to investigate steam condensation inside tubes in the presence of non-condensable gases. The work was independently conducted at the University of California at Berkeley and at the Massachusetts Institute of Technology (MIT). The work was initiated to obtain a database and a correlation for heat transfer and condensation inside tubes. Three researchers utilized three separate experimental configurations at the University of California at Berkeley, while two researchers utilized one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and non-condensable mass fractions. The researchers found the system well-behaved for all tests with either of the non-condensable gases. The results of the tests at the University of California at Berkeley are the basis for the condensation heat transfer correlation used in the TRACG computer code.

TRACG ICS modeling is qualified by the PANTHERS IC test using a representative configuration. The steady-state heat exchanger performance was predicted by the PANTHERS IC prototypical geometry full-scale test.

Because the BWRX-300 RPV and ICS are similar to those of the ESBWR, the TRACG method developed for the ESBWR RPV thermal-hydraulics and mass energy release is also used for the BWRX-300 RPV thermal-hydraulics and mass and energy release. The TRACG code and the application method developed for ESBWR was reviewed and approved by the USNRC. That application method was developed using the Code Scaling, Applicability and Uncertainty (CSAU) guidance. GEH Licensing Topical Report NEDC-33922P BWRX-300 Containment Evaluation Method (Reference 15.5-2) provides an overview of the TRACG thermal-hydraulics method for the mass and energy release and its applicability to the BWRX-300 RPV.

#### **15.5.1.2.2 GOTHIC**

Containment analysis is performed by using the GOTHIC code (References 15.5-4 and 15.5-5).

The GOTHIC computer code is a state-of-the-art program for modeling multiphase, multicomponent fluid flow for performing both containment DBA analyses and analyses to support equipment qualification. The GOTHIC code is developed by Numerical Applications Incorporated (NAI), and the development program is sponsored by the Electric Power Research Institute (EPRI).

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The GOTHIC code has a nodding structure that allows both lumped parameter and 3-D modeling capabilities. The multi-dimensional analysis capability facilitates the study of non-condensable gas and stratification and the calculation of flow field details within any given volume. The code has undergone extensive review and validation against a large test array. The validation program scope examines the code capability for predicting pressure and temperature as well as hydrogen distribution and mixing under various conditions.

GOTHIC is a continuously maintained and improved computer code. The GOTHIC code has been developed compliant with US Title 10 Code of Federal Regulations Part 50, Appendix B, Quality, meeting the GEH software quality requirements and complies with REGDOC-2.4.1, Section 4.4.5 / N286.7 (Reference 15.5-6) requirements. The PSAR results generated used GOTHIC Version 8.3 that is the latest released version. Future BWRX-300 containment analyses may be performed using newer versions of the GOTHIC code provided the newer versions meet the same USNRC 10 CFR 50, Appendix B quality and REGDOC-2.4.1, Section 4.4.5 / N286.7 requirements, as well as changes in calculated results for the BWRX-300 containment application caused by any code changes that can be successfully dispositioned.

#### **15.5.1.2.3 PANAC11**

The BWR Core Simulator (PANAC11 or P11) is a steady-state, 3-D coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop. An Automated Plant Heat Balance (APHB) is available to model an external flow loop. Provisions are made for fuel cycle and thermal limits calculations. Neutronic parameters used by core simulator are obtained from two-dimensional lattice physics and parametrically fitted as a function of moderator density, exposure, control, and moderator density history for a given fuel type. The simulator is used for detailed 3-D design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A special power exposure iteration option is available for target exposure distribution and cycle length predictions. PANAC11 includes the effect of Doppler broadening as a function of moderator density, exposure, control, and moderator density history for a given fuel type. The lattice physics and core simulator form the nuclear simulator and is used for both core design and core exposure tracking.

The nuclear model is based on a coarse-mesh nodal, 1-1/2 group (quasi-two group), static diffusion theory. The diffusion equations are solved using the fast energy group. Resonance energy and thermal energy neutronic effects are included in the model by relating the resonance and thermal energy fluxes to the fast energy flux. Eigenvalue iteration yields the fundamental mode solution. Control blade history local peaking effects are considered. A pin power reconstruction model is incorporated to account for the effect of flux gradients across the nodes on the local peaking distribution. Instrumentation predictions are made for neutron and gamma sensitive detectors.

Neutronic parameters used by PANAC11 are obtained from the 2-D lattice physics code TGBLA06, as described in the preceding section, and parametrically fitted as a function of moderator density, exposure, control, and moderator density history for a given fuel type.

The nuclear model is coupled to a static, parallel channel, thermal-hydraulics model and is consistent with the approach described in the Global Nuclear Fuel (GNF) BWR steady-state thermal-hydraulic method NEDC-32082P (Reference 15.5-7).

The GE BWR Core Simulator has undergone extensive validation by comparing calculated results with alternate 3-D methods, end of cycle gamma scan data, operating reactor data, and transient experimental data.

PANAC11 is an USNRC-approved method used for production core design and licensing analysis for the fleet of BWRs with GNF fuel. TGBLA06 and PANAC11 form the nuclear simulator and are used for both core design and core exposure tracking. This tracking process is a continuous comparison and validation of the nuclear methods descriptions, a demonstration of methods performance through operational qualification through similar reactors for the BWRX-300, and the quantification of uncertainties that are applicable in thermal margin evaluations that are a function of this performance are provided in NEDC-33939P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," (Reference 15.5-16).

#### **15.5.1.2.4 ANSI/ANS-18.1-2020**

Radiation concentrations in the reactor coolant and steam during normal operations are determined based on ANSI/ANS-18.1-2020 (Reference 15.5-10). This standard provides the bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR plant. The source term data is based on the cumulative industry experience at operating BWR plants, including measurements at several stations. The operating data reflects the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs such as the GNF2 fuel used in the BWRX-300.

#### **15.5.1.2.5 ADDAM**

The ADDAM (Atmospheric Dispersion and Dose Analysis Method) computer code (Reference 15.5-8) computes the statistical distribution of radiation doses to an individual or population after the airborne release of radioactive material into the environment following a design-based nuclear accident at a nuclear facility. The dispersion of the release is highly influenced predominately by different characteristics of release, existing meteorological scenarios, and the overall nearby terrain and building dimensions. Doses can be computed for various age groups, organs, and receptor locations which are classified based on release and exposure pathway. Input data into ADDAM is divided into four sections:

- Meteorological conditions
- Site characteristics
- Release characteristics
- Receptor characteristics

The execution of the ADDAM code is accomplished by the following input data files:

- Meteorological
- Radionuclide
- Release activity
- Stack exit temperature and velocity file
- Release thermodynamic data files

ADDAM complies with the requirements of CSA N288.2-19 (Reference 15.5-9) standard and is the current CANDU Owners Group's Industry Standard Toolset Version for analyses. CSA N288.2-19, Clause 1.5 states that the "standard covers local atmospheric dispersion, which for Gaussian plume models is defined as dispersion that occurs in the range of 300 m to 100 km." While it does not explicitly preclude use below exclusion zone radii below 300 m, it does mention the need for specialized models. The application within is the current recommendations of the standard as implemented in ADDAM. The individual public doses, dilution factors ( $\chi/Q$ ), and deposition factors ( $D/Q$ ) at a 350m exclusion zone boundary radius for a 95th percentile cumulative frequency of occurrence cut-off have been calculated using a modified reference calculation and output options is used for all the production runs for dilution factors and dose assessment as these are standard output options of ADDAM.

### **15.5.2 Analysis of Normal Operation**

The normal operation DSA demonstrates that plant parameters are maintained within specified Operating Limits and Conditions (OLCs) ensuring the plant conforms with the safety analysis assumptions. The main objective of the first defence level of a D-in-D strategy is to prevent challenges to plant equipment and to protect the primary physical barriers – fuel cladding and RCPB. DL1 measures are described in Chapter 3, Section 3.1.6.2.

The measures used to limit radiological releases to the public during normal operations are described in Chapter 11 Management of Radioactive Waste Systems, Chapter 12 Radiation Protection, and Chapter 20 Environmental Aspects.

#### **15.5.2.1 Description of Normal Operational Modes**

Normal operation is defined as operation within specified OLCs. An analysis of normal operation includes all operational modes (Modes 1-6 defined in Chapter 16, Appendix 16A).

#### **15.5.2.2 Method and Scope of Analysis**

The normal operation deterministic safety analysis demonstrates that the plant parameters are maintained within the specified OLCs ensuring that the plant conforms with the safety analysis assumptions. The normal operation of the plant is monitored and controlled so that PIEs that may lead to AOOs are mitigated before evolving into DBAs. OLCs are important DL1 measures that are readily checked by the operators ensuring the facility is operated in accordance with the applicable safety analyses. OLCs define minimum levels and allowable configurations of the plant, equipment, and associated resources needed to enact safety measures. Setpoint limits and conditions define where safety measures are intended to be activated or initiated to protect against or mitigate fault sequence consequences.

Source term analysis is provided in Chapter 11, Section 11.1 for normal operation sources. Radiation protection measures analyzed and implemented for the plant, shielding and radiation zone designation is provided in Chapter 12, Sections 12.4 and 12.5, respectively.

#### **15.5.2.3 Results of Analysis**

The limiting safety system setpoints are determined to demonstrate to the extent practicable that the highest safety class (DL3) systems are only initiated when needed. Process controls and alarms are established that demonstrate their effectiveness in reducing or avoiding the need for safety system actuations. The core parameters evaluation ensures that reactivity control required to bring the reactor to a cold shutdown condition is achieved and maintained. Normal operational doses are provided in Chapter 20, Section 20.8.

#### **15.5.2.4 Stability Analysis**

Part of normal operational analysis is to confirm that the core will remain stable during normal operation. The stability considerations during normal operation are described in Chapter 4, Section 4.8.

The BWRX-300 240-bundle core is evaluated for beginning of cycle (BOC), middle of cycle (MOC) and end of rated power (EOR) exposures in determining both the core-wide decay ratios and regional mode oscillations. The core-wide decay ratios (DR) are evaluated using a step perturbation in pressure while the regional mode decay ratios are evaluated using channel velocity perturbations.

##### **Core-Wide Decay Ratio**

The primary stability evaluation is performed at nominal conditions including nominal feedwater temperature 241.9°C (467.4°F).

Another stability evaluation is performed for the state that is reached after a LFWH AOO. A Select Control Rod Run-In (SCRRI) (described in Chapter 7, Subsection 7.3.3.1) is initiated as a mitigating response for a LFWH AOO with a feedwater temperature reduction of 16.7°C (30.0°F) or higher and a new lower power steady condition is achieved. Once the reactor achieves a new steady-state condition, a step pressure perturbation is applied to evaluate core stability response. The analysis of LFWH AOO described in Section 15.5.3.1.1 assumes that the feedwater temperature is reduced to 191.9°C (377.4°F) at BOC, MOC and EOR.

##### **Core-Wide Dominance**

For the regional mode evaluation, based on the harmonic modes distribution of the core, the inlet velocities for all channels were perturbed by  $\pm 20\%$  at time = 0. This harmonic power distribution is predicted by the steady-state core simulator PANAC11 and results in a line of symmetry between the two halves of the core with higher and lower predicted harmonic power. The velocity perturbations are made positive on one side of the line of symmetry and negative on the other side. This stimulates the potential harmonic oscillations (regional oscillations). The resulting channel power response of limiting channels is evaluated for susceptibility to regional mode oscillations. If the core is not susceptible to regional mode oscillations after a velocity perturbation, the initially symmetric, out of phase channel power responses come into phase after a short duration, confirming the dominance of core-wide oscillations.

#### **Results**

##### **DR / SCRRI**

The maximum nominal core-wide decay ratio design limit is 0.80. The calculated core-wide decay ratios at nominal conditions are below the maximum decay ratio allowed. The calculated DR values at the end of the LFWH event are below the maximum allowed DR and are lower than the DR values at nominal conditions. The calculated DR values at nominal temperature and LFWH conditions are presented in Table 15.5-2. The nominal stability response is presented on Figure 15.5-1 for the MOC exposure. The LFWH stability response is presented on Figure 15.5-2 for the MOC exposure.

## Core-Wide Dominance

A limiting core-wide dominance evaluation is performed at 115% rated power. At MOC, the limiting channel power values follow the expected behavior where initially symmetric, out of phase channel power responses come into phase after a short duration. The core is not susceptible to regional mode oscillations at nominal conditions, and this conclusion also applies to normal operation. The regional stability response using the limiting channels is presented on Figure 15.5-3 at the MOC exposure. This is a hypothetical evaluation, and any growing core-wide oscillations are mitigated by DL3-05 (see Table 15.5-5).

Based on these stability evaluations, the following stability claims are supported:

1. Power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible
2. Regional instability is not possible
3. Design features prevent the loss of stability margin for upset events

## 15.5.3 Analysis of Anticipated Operational Occurrences

### 15.5.3.1 Decrease in Core Coolant Temperature AOO

This section describes the bounding BL-AOO event for Temperature Decrease (TD) Group.

#### 15.5.3.1.1 Loss of Feedwater Heating AOO

This event is designated as a BL-AOO event. The fault sequence name is Loss of Feedwater Heating (LFWH) and the fault sequence ID is TD-LFWH\_BL-AOO.

Additional LFWH AOO cases support the detailed evaluation and demonstrate the thermal-hydraulic stability of the BWRX after a LFWH AOO (see 15.5.2.4).

## Postulated Initiating Event

The event assumes a loss of FW heating from a single failure of either the closure of one extraction steam valve or the inadvertent bypass of a FW heater. This failure is conservatively modeled as an instantaneous decrease in FW temperature that bounds the maximum FW temperature decrease resulting from a single failure. A PIE with AOO frequency results in the maximum FW temperature reduction identified in Table 15.5-3 from a loss of one feedwater heater.

## Sequence of Events

The event fault sequence summary:

- Loss of FW temperature occurs instantly resulting in an increase in power
- Reactor Level Control (RLC) compensates initially by lowering flow rate minimizing the effect on power
- SCRRRI inserts control rods on indication of FW temperature reduction
- Reactor Pressure Control (RPC) maintains pressure and RLC maintains level
- A new controlled steady-state condition achieved with a new power distribution

Table 15.5-6 lists the sequence of events.

## Identification of Operator Actions

No operator action is required to mitigate the event.

## **Systems Operation**

Credited DL2 functions:

- DL2-27 – SCRR1 on FW Temperature Decrease
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

## **Core and System Performance**

### *Input Parameters and Initial Conditions*

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

### *Results*

The results of the simulated loss of FW heating event are presented on Figures 15.5-5 through 15.5-10. The results are presented in Table 15.7.2-1 for the exposure with the limiting CPR response.

The reduced temperature FW enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. RLC compensates initially by lowering FW flow rate and a SCRR1 is initiated to minimize the core power increase and decrease the final steady-state power. Steam flow and FW flow then stabilize at a lower level. The RPV water level decreases and then returns to normal level. The pressure and level remain well within the RPV water level and RCPB pressure acceptance criteria in Section 15.3.2.

The core power increase is limited. Thermal-mechanical evaluations confirm there is significant margin to centerline fuel temperature or cladding strain acceptance criteria in Section 15.3.2. Limits on the Linear Heat Generation Rate (LHGR) are included each operating cycle ensuring the centerline fuel temperature and cladding strain acceptance criteria are met. The limits on LHGR are included in the Core Operating Limits Report (COLR).

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Sensitivity studies were performed on maximum FW pump flow (120% of rated), initial FW temperature reduction ( $-6^{\circ}\text{C}$ ), and FW controller settings. The sensitivity studies demonstrated no significant change in the event sequence or results.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.



### **15.5.3.2 Increase in Reactor Pressure AOO**

#### **15.5.3.2.1 Generator Load Rejection or Turbine Trip AOO**

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Generator Load Rejection or Turbine Trip (LR-TT), and the Fault Sequence ID is PI-LR-TT\_BL-AOO.

#### **Postulated Initiating Event**

The initiating event is either a generator load rejection or a turbine trip. TCVs have a fast closure function to protect the turbine during a generator load rejection. The TSVs close at a fast rate following a turbine trip. RPC remains unaffected and demands the TBVs open to control reactor pressure.

#### **Sequence of Events**

The fault sequence summary for the event:

- TCVs and/or TSVs close quickly causing pressure increase
- Anticipatory scram occurs on indication of a load rejection or turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved

Table 15.5-7 lists the sequencing of events for this analysis.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

#### **Systems Operation**

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

#### **Core and System Performance**

##### *Input Parameters and Initial Conditions*

The event is simulated by initiating a generator load rejection or turbine trip resulting in a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated generator load rejection / turbine trip is presented on Figures 15.5-11 through 15.5-16, and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR result. Automatic reactor scram occurs following indication of a generator load rejection or turbine trip. Pressure increases but is limited by the TBVs opening.

The core thermal power does not increase above the initial power and there is no concern in approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR Acceptance Criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.

#### **15.5.3.2.2 Closure of One Main Steam Reactor Isolation Valve AOO**

This event is in the Pressure Increase (PI) Group and is designated as a BL-AOO event. The fault sequence name is Closure of one Main Steam Reactor Isolation Valve (MSRIV) and the Fault Sequence ID is PI-1MSRIVC\_BL-AOO.

### **Postulated Initiating Event**

There are two main steam lines. The event is an inadvertent closure of one MSRIV that terminates flow in one of the main steam lines. A minimum MSRIV closure time results in the most severe event.

### **Postulated Event**

The fault sequence summary for the event:

- One MSRIV closes causing RPV pressure and power to increase
- ATS scram occurs on MSRIV position
- RLC controls levels
- Second MSRIV in the second steam line closes on leak detection indication (this is assumed because it makes the event more severe)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-8 lists the sequencing of events for the closure of one MSRIV.

### Identification of Operator Actions

No operator action is required to mitigate the event.

### Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-21 – Anticipatory Hydraulic Scram on MSRIV/MS CIV Position
- DL2-31 – ICS Pressure Control on High Reactor Pressure

### Core and System Performance

The event is simulated by initiating a closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The simulated closure of one MSRIV is presented on Figures 15.5-17 through 15.5-22 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR and reactor pressure response. The neutron flux and pressure increase resulting from the closure of one MSRIV are limited by an anticipatory scram on MSRIV position. The pressure increase is also initially limited because the MSRIV remains open in the second steam line. The second steam line is then assumed to close on main steam line break indication. This conservative assumption makes the event more severe. Pressure then increases and ICS is initiated on high pressure.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

#### *Barrier Performance*

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.3.2.3 Loss of Condenser Vacuum AOO**

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Loss of Condenser Vacuum (LOCV) and the Fault Sequence ID is PI-LOCV\_BL-AOO.

### **Postulated Initiating Event**

There are a few potential causes of a loss of condenser vacuum including loss of one or more circulating water pumps. The loss of condenser vacuum results in a turbine trip. The turbine stop valves close at a fast rate following a turbine trip.

### **Sequence of Events**

The fault sequence summary for the event:

- The TSVs close and main turbine trips on low main condenser vacuum causing pressure increase
- Anticipatory scram occurs on a turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- TBVs close on high main condenser pressure and pressure increases slowly due to decay heat (the simulation is ended before TBV closure because the key mitigation DL functions are demonstrated, and a single ICS train is capable of controlling pressure and removing decay heat)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-9 lists the sequencing of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-31 – ICS Pressure Control on High Reactor Pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The loss of condenser vacuum results in a turbine trip. The event is simulated by initiating a turbine trip resulting in a fast TSV closure. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

#### *Results*

The simulated loss of condenser vacuum is presented on Figure 15.5-23 through 15.5-27 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR result. The pressure increases due to the fast TSV closure and is limited by the anticipatory scram on indication of a turbine trip. A scram on high main condenser pressure may occur sooner but conservatively not modeled. The pressure increase is also initially limited by the TBVs opening. Condenser vacuum loss is assumed to continue, resulting in TBVs closing. Once TBVs are closed, reactor pressure increases, and one ICS train initiates on high RPV pressure.

The core thermal power does not increase above the initial power and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development.

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.3.2.4 Loss-of-Preferred Power AOO**

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Loss-of-Preferred Power (LOPP) and the Fault Sequence ID is PI-LOPP\_BL-AOO.

#### **Postulated Initiating Event**

A LOPP is initiated by offsite power supply failure. The loss of power results in the generator output breakers opening and the TCVs fast closure.

#### **Sequence of Events**

The fault sequence summary for the event is:

- LOPP occurs
- TCVs close quickly causing pressure increase
- Feedwater pumps lose power, FW pump discharge check valves maintain coolant inventory
- Circulating water pumps lose power
- Anticipatory scram occurs on generator load rejection
- RPC opens TBVs to control pressure

- TBVs close on high main condenser pressure
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-10 lists the sequencing of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL2 functions:

- DL2-01 – Maintain Target Pressure
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-31 – ICS Pressure Control on High Reactor Pressure
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The LOPP results in the generator output breakers opening and a loss of power to the feedwater pumps. The event is simulated by initiating a FW trip and a load rejection resulting in fast TCVs closure. Anticipatory scram occurs on a generator load rejection. A scram on low bus voltage may occur sooner but is conservatively not modeled. This scram timing is the same as the anticipatory scram on a load rejection. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions. The results are shown for the case with the limiting result for CPR.

#### *Results*

The simulated LOPP is presented on Figure 15.5-28 through 15.5-33 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting result for CPR. The pressure increase due to TCV closure is limited by the anticipatory scram on the generator load rejection. Scram on a low electric bus voltage may occur sooner but is conservatively not credited. The pressure increase is also initially limited by the TBVs opening. The TBVs later close on loss of power. Once the TBVs are closed, reactor pressure increases and ICS initiates on high RPV pressure. The ICS continues to limit the pressure increase.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

### **15.5.3.3 Decrease in Reactor Coolant Inventory AOOs**

#### **15.5.3.3.1 Feedwater Pump Trip – One Pump**

The section analyzes the bounding BL-AOO event for the Inventory Reduction (IR) group. The fault sequence name is FW Pump Trip – One Pump (FWPT) and the Fault Sequence ID is IR-FWPT\_BL-AOO.

#### **Postulated Initiating Event**

There is one FW pump normally operating and a second FW pump in standby. This event assumes a failure resulting in a trip of the operating FW pump. The RLC remains unaffected by the failure and increases the flow demand on the standby FW pump to maintain RPV water level.

#### **Sequence of Events**

The fault sequence summary for the event:

- One FW trips causing RPV water level decrease
- Standby FW pump starts and increases to rated FW flow
- Power decreases temporarily from a reduction in core flow and core inlet subcooling
- RPC controls pressure
- RLC maintains level
- RPV water low level scram and high-level FW isolation are avoided
- Controlled state achieved

Table 15.5-11 lists the event sequence.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

#### **Systems Operation**

Credited DL2 functions:

- DL2-25 – Start Standby FW pump on Loss of Operating FW Pump
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

## Core and System Performance

### *Input Parameters and Initial Conditions*

The event is simulated by initiating a FW pump trip. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The FW pump trip event is presented on Figures 15.5-34 through 15.5-39 and the results are presented in Table 15.7.2-1 for the exposure with the limiting CPR response. Reduction in FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The standby FW pump starts on confirmed low FW flow conditions, RPC throttles TCVs to control pressure, and the RLC increases the FW pump flow to rated conditions to maintain level. Low reactor water level (L3) scram is avoided.

The core thermal power does not increase and there is no concern for in approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3.2. This event is not limiting and is not considered during development of LHGR limits.

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. The event is not limiting and is not considered in the OLMCPR development.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

## **15.5.3.4 Increase in Reactor Coolant Inventory AOOs**

### **15.5.3.4.1 Inadvertent Isolation Condenser Initiation – One Train**

This event is designated as a BL-AOO event. The fault sequence ID is II-IIICI-1\_BL-AOO. This event assumes a failure causes a single Isolation Condenser (IC) condensate return valve to open. The event assumes that RLC remains unaffected by the failure and is able to maintain level. The event also assumes that RPC remains unaffected by the failure and is able to maintain pressure.

### **Postulated Initiating Event**

The ICs are normally in standby mode. This event assumes spurious opening of a single ICS condensate return valve, resulting in the introduction of cold water into the reactor. The event assumes that RLC and RPC remain unaffected by the failure and are available to control reactor level and reactor pressure.

### **Sequence of Events**

The fault sequence summary for the II-IIICI-1\_BL-AOO event:

- ICS condensate return valve on one train opens
- Cold ICS condensate water drains into the chimney



- RLC maintains water level
- RPC maintains pressure
- Controlled state achieved

Table 15.5-12 lists the sequencing of events for the feedwater flow increase event.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

The credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

### **Input Parameters and Initial Conditions**

The event is simulated by opening the ICS condensate return valve on one IC train. The initial conditions are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### **Results**

The inadvertent initiation of one IC train event is presented on Figures 15.5-40 through 15.5-45. Table 15.7.2-1 shows the limiting results for CPR response. When the IC condensate return valve is opened, cold water is introduced into the chimney region. After an initial small perturbation, the increased density in the chimney reduces core flow, water level, and power temporarily. After the initial reduction, core flow, level, and power increase. When the increase in level is sensed, the FW controller starts to demand the operating FW pump to reduce flow. After the initial surge, as condensate water drains into the chimney and IC flow reduces, the FW controller demands the operating FW pump to increase flow. The RPV water level, core flow, and core power settle back to their initial values. RPV pressure increases insignificantly. The level and pressure remain well within the RPV water level and RCPB pressure acceptance criteria in Subsection 15.3.2.

The core power increase is limited. There is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.2. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated  $\Delta\text{CPR}/\text{ICPR}$  is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

### **Barrier Performance**

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

### **Radiological Consequences**

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4 Analysis of Design Basis Accidents**

This section evaluates the bounding BWRX-300 non-LOCA and LOCA PIEs. Subsections 15.5.4.1 through 15.5.4.5 describe the DSA non-LOCA DBAs, while Section 15.5.4.6 describes the DSA LOCAs inside containment. Subsection 15.5.9.2 describes the DSA for LOCAs outside containment.

##### **15.5.4.1 Decrease in Reactor Coolant Temperature Event**

###### **15.5.4.1.1 Loss of All Feedwater Heating**

This event is in the Temperature Decrease (TD) group and is designated CN-DBA event. The fault sequence name is CCF-LFWH, Passive Digital CCF DL2 Technology Platform (CCF-DL2), and the Fault Sequence ID is TD-CCF-LFWH\_CCF-DL2\_CN-DBA.

##### **Postulated Initiating Event**

A CCF results in the loss of all FW heating. Any CCF that results in the loss of all FW heating occurs gradually because of the thermal inertia inherent in the FW heaters. The FW temperature lowers to the main condenser temperature with the assumed time constant shown in Table 15.5-3.

##### **Sequence of Events**

- CCF results in the loss of all FW heating
- FW temperature decreases causing positive reactivity insertion
- RLC and RPC fail as-is
- Scram on high Simulated Thermal Power (STP) causing negative reactivity insertion
- RPV pressure decreases. The downcomer level decreases temporarily to lower than L3 because of void collapse
- Main steam isolation occurs on low RPV pressure
- RLC continues at initial flow causing RPV level to increase
- FW isolation occurs on high RPV level
- An ICS initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining ICS trains sufficiently controls pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-13 lists the sequence of events.

##### **Identification of Operator Actions**

No operator action is required to mitigate the event.

##### **Systems Operation**

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level

- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

## **Core and System Performance**

### *Input Parameters and Initial Conditions*

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated loss of all FW heating event is presented on Figures 15.5-46 through 15.5-51, and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The reduced FW temperature enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient, and a scram occurs on high STP. MSRIV isolate on low RPV pressure.

FW flow remains at 100% due to the RLC CCF and RPV water level rises until FW isolates on high RPV water level. Decay heat causes RPV pressure to rise and an ICS train initiates. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train starting on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and the RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

### *Barrier Performance*

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

## **15.5.4.2 Increase in Reactor Pressure Events**

### **15.5.4.2.1 Generator Load Rejection or Turbine Trip**

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is Generator Load Rejection or Turbine Trip (LR-TT), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is PI-LR-TT\_CCF-DL2\_CN-DBA.

### **Postulated Initiating Event**

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is, which are continually operating, and failure of the anticipatory scram.

### **Sequence of Events**

The fault sequence summary for the event is as follows:

- TCVs and/or TSVs close quickly causing pressure and power increase
- RLC fails as-is at initial condition, the TBVs remain closed, and anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- RPV pressure continues to increase because RPC fails as-is
- RPV level reduces due to the pressure increase
- One ICS train initiates on high pressure. First IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- With RLC failing as-is, initial feedwater flow continues causing RPV level to increase
- FW isolates on high RPV water level (L9)
- Controlled state achieved

Table 15.5-14 lists the sequencing of events, for the fast closure of TCV.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High neutron flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by initiating a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. This event is limiting for PCT. Therefore, additional initial conditions and phenomena that impact PCT described in the TRACG Application (Reference 15.5-3) are applied in the limiting direction by at least one standard deviation consistent with the approach for an event with medium margin:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- Critical quality used in boiling transition correlation
- Channel radial peaking factor
- Hot rod power
- Total initial power

Analysis is performed as needed to confirm the conservative PCT direction. Then a bounding case is created with all inputs in the conservative direction.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The simulated generator load rejection / turbine trip is presented on Figures 15.5-52 through 15.5-57 and the results are presented in Table 15.7.3-1. The results are presented for the bounding case described above. Reactor scram occurs following high neutron flux. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. TBVs fail to open; however, ICS limits the pressure increase. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling. A single failure of an ICS train to start on high pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria, provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Applying these conservatisms have little effect on peak pressure because peak pressure is primarily driven by the IC initiation setpoint. Once the pressure setpoint is reached, ICS initiates, and pressure rapidly reduces. For PCT, these uncertainties result in a PCT 91°C (163°F) higher than the base case with conservatisms similar to other DBA analyses. The results are within the acceptance criteria.

#### *Barrier Performance*

The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.2.2 Loss-of-Preferred Power**

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is LOPP, and the Fault Sequence ID is PI-LOPP\_CCF-DL2\_CN-DBA.

#### **Postulated Initiating Event**

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is at the initial condition, which are continually operating, and failure of the anticipatory scram.

## Sequence of Events

The fault sequence summary for the LOPP event:

- TCV closes slowly due to loss of turbine control hydraulic pumps
- FW pumps lose power and coast down
- RLC fails as-is at the initial condition, TBVs remain closed, and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- RPV pressure continues to increase because TBVs remain closed
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the remaining two trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-15 lists the sequencing of events.

## Identification of Operator Actions

No operator action is required to mitigate the event.

## Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-12 – ICS Train 2 initiation on High RPV Pressure
- DL3-39 – FW Isolation on Loss of Normal FW Flow

## Core and System Performance

### *Input Parameters and Initial Conditions*

The event is simulated by initiating a slow closure of the TCVs and feedwater pump trip. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated LOPP is presented on Figures 15.5-58 through 15.5-63 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. Reactor scram occurs following high neutron flux. Bypass valves fail to open; however, the pressure increase is limited by the ICS initiation. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling; therefore, a single failure of an ICS train to start on high pressure does not affect the event mitigation. The closure of the TCVs in the LOPP AOO is due to DL2 active mitigation. In the LOPP CN-DBA sequence, the TCVs close as a result of the PIE. If there is no power to maintain hydraulic pressure, the valves slowly close.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient challenge to the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.2.3 RPV Pressure Control Downscale**

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is CCF – RPV Pressure Control Downscale (CCF-RPCD), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is PI-CCF-RPCD\_CCF-DL2\_CN-DBA.

#### **Postulated Initiating Event**

The PIE is a spurious CCF of the RPV pressure control. This failure results in a demand to close the TCVs (normal servo closure). This PIE also prevents the TBVs from opening. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in the RLC failing as-is at initial conditions, and failure of the anticipatory scram.

#### **Sequence of Events**

The fault sequence summary for the RPCD event:

- RPC demands TCVs to slow close and the TBVs remain closed
- RLC fails as-is at the initial condition and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- With RLC failing as-is at the initial condition, the initial feedwater flow continues causing RPV level to increase
- FW isolates on high RPV water level
- One ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining ICS trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-16 lists the sequencing of events.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

## **Systems Operation**

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV pressure

## **Core and System Performance**

### *Input Parameters and Initial Conditions*

The event is simulated by initiating a slow closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated RPV Pressure Control Downscale is presented on Figures 15.5-64 through 15.5-69 and the results are presented in Table 15.7.3-1. The results are shown with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. Reactor scram occurs following high neutron flux. TBVs fail to open, and feedwater pump trips on high RPV level (L9). Natural circulation continues at a rate consistent with decay heat power. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling; therefore, a single failure of an ICS train to start on high pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.2.4 Closure of All Main Steam Reactor Isolation Valves and FW Isolation Valves**

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is CCF-Closure of All MSRIVs and FW isolation valves (CCF-DL4a-MSRIVC-FWIV) and the Fault Sequence ID is PI-CCF-DL4a-MSRIVC-FWIV\_CN-DBA.



### **Postulated Initiating Event**

The PIE is a spurious CCF DL4a function that affects all MSRIV and FW isolation valves.

### **Sequence of Events**

The fault sequence summary for this MSRIV and FWIV closure event:

- Closure of all MSRIV and FW isolation valves
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-17 lists the sequence of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL3 functions:

- DL3-04 - Hydraulic Scram on High neutron flux
- DL3-12 – ICS Train 2 initiation on High RPV pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by initiating a closure of the MSRIV and FW isolation valves. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The simulated closure of MSRIVs and FW isolation valves is presented on Figures 15.5-70 through 15.5-75 and the results are presented in Table 15.7.3-1. The results are shown for the limiting case for cladding temperature and reactor pressure response. Reactor scram occurs following high neutron flux. The pressure increase is limited by ICS initiation. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling. A single failure of an ICS train to start on high pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient that challenges the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.

### **15.5.4.3 Reactivity and Power Distribution Anomalies (DBA)**

This event group is in the Reactivity Increase (RI) group.

#### **15.5.4.3.1 Fuel Loading Error Event**

The event is designated as a CN-DBA event. The fault sequence name is Fuel Loading Error (FLE) and the Fault Sequence ID is RI-FLE\_CN-DBA.

Both the mislocated and the misoriented fuel bundle events are referred to as the fuel loading error event. For a fuel loading error event, it is assumed that the improper loading of a fuel assembly is not discovered and corrected through the core verification program, and the plant is operated throughout the operating cycle assuming that the design core configuration has been correctly implemented.

#### **Mislocated Fuel Bundle**

The mislocated fuel bundle error involves the loading error of two fuel bundles-the misloaded bundle and the bundle that belongs in that location. The scenario includes: 1) one location loaded with a bundle that operates at a lower power than planned and 2) another location with a bundle operating at a higher power than planned.

Three errors must occur for the mislocated event to take place:

1. Bundle is misloaded into a wrong core position.
2. Fuel bundle is loaded in the location where the first mislocated bundle occurred is overlooked, resulting in another mislocated bundle.
3. Both misplaced bundles are overlooked during the core verification performed following core loading.

The misoriented fuel bundle error involves the loading error of one fuel bundle to be rotated 180 degrees from the intended orientation.

For the misorientation event, two errors must take place:

1. The assembly is rotated while being lowered into position.
2. The misoriented bundle is overlooked during the core verification performed following core loading.

It is assumed that the FLE is not detected and that fuel rods operate above the thermal and mechanical limits. The potential exists that one or more fuel rods experience cladding failure. If this were to occur, the adverse consequences are detectable and can be suppressed during operation similar to leaking fuel rods resulting from other failure mechanisms. For the FLE, the initial adverse consequences consist of perforation of a small number of fuel rods in the assembly. Any perforations in the fuel cladding that occurs is localized and does not propagate to other

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

assemblies. A control rod inserted in the vicinity of the leaking fuel rod(s) suppresses the power in the leaking fuel rod(s) and reduces the fission product release and offgas activity.

The FLE may exceed the operating mechanical LHGR limit because it may have worse peaking than the normally-loaded bundle. If the FLE operates above the operating mechanical LHGR limit, one or more rods may approach the design limit and experience cladding failure. If this occurs, the adverse consequences are the perforation of a small number of fuel rods in the misplaced bundle. The subsequent release of fission products to the reactor coolant is detected by the offgas system and processed accordingly.

### **Sequence of Events**

- Multiple layers of independent administrative controls barrier due to human actions fail
- Startup occurs with a bundle rotated or with two bundles swapped locations
- Bundle swap or rotation results in potential for limited fuel cladding failures
- Rod failures (if they occur) occur over an extended period of time
- Elevated radiation in reactor coolant

### **Identification of Operator Actions**

No operator action is credited for mitigating this event.

### **Systems Operation**

No mitigation functions are credited in this event.

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The core operates at normal conditions during the cycle.

#### *Results*

It is possible that operators recognize the mislocated or misoriented bundle during startup or initial core operation using operating procedures because local monitoring indicates the reactivity anomaly. This allows the reactor operators to mitigate the event, which is not credited in the analysis. It is also possible that the mislocated or misoriented bundles do not result in significant local reactivity changes or fuel cladding failures.

If there is a significant increase in local reactivity, then the fuel could operate above LHGR limits or enter boiling transition. Analysis shows that the possibility of entering boiling transition could occur in relatively hot rods and in one of the mislocated or misoriented bundles and potentially the four face adjacent bundles. The mislocated bundle bounds the misoriented bundle for the potential to increase the local reactivity and result in failed fuel rods.

Even if some rods did enter boiling transition or operate above the LHGR limits, the rod failures occur over an extended time (weeks or months) or fail intermittently. Rod failures may be recognized as essentially fuel leakers. Operators would observe the increase in offgas system activity and take appropriate actions. Once recognized, procedures require the operators to locate the failed rods and, once located, insert a control rod in the location to suppress the power. If power suppression is not successful, the reactor is shutdown. Neither the slow developing fuel failures nor the operator actions are credited in the radiological analysis.

### *Barrier Performance*

The reactor remains at normal operating pressure throughout the event. There is no challenge to the RPV and primary systems such as process barrier stress limitations.

The containment also remains at normal operating pressure and temperature. Containment integrity is not challenged.

As discussed above, there is potential for fuel failure. Analysis of potential mislocated bundles indicates that the number of fuel rod failures is bounded by assuming all the fuel rods in five bundles fail.

### *Radiological Consequences*

A conservative approach is used to evaluate the radiological consequences. Instead of one or two rods failing, the event is analyzed for all the fuel rods in a mislocated or misoriented fuel assembly, and all the rods in the mislocated bundle and the four adjacent fuel assemblies experience instantaneous failure during normal operation.

To further assure that the fuel bundles contain the maximum fission product release, all five bundles (array independent) are multiplied by:

1. A factor of 1.4 to account for variations in fission product inventory over the operational cycle
2. A second factor of 2.5 to account for variations in cycle-dependent bundle power as a ratio to the end of cycle average bundle power

This results in a total factor of  $1.4 \times 2.5 = 3.5$  to bound the bundle end of cycle inventory.

The radiological consequences of failing all of the fuel rods in five fuel bundles have been analyzed to provide results for no isolation trip with the release treated by an augmented offgas system.

The result for this analysis is presented in Table 15.7.3-2.

The radiological consequence of failing all fuel rods in five fuel bundles was analyzed for the case where the plant does not have a main steam line high radiation isolation trip. With no automatic MSIV closure, the activity is transported to an offgas system. The activity release to the environment would occur from the normal offgas release point after holdup in the offgas treatment system.

The results listed in Table 15.7.3-2 demonstrate that the radiological consequence do not exceed the offsite dose requirements for DBAs. The fuel loading error event is not a limiting event for DBA criteria for RCPB or containment integrity because the reactor remains at normal operating pressure throughout the event. There is no challenge to the RPV and primary systems, such as process barrier stress limitations. Containment also remains at normal operating pressure and temperature.

#### **15.5.4.4 Increase in Reactor Coolant Inventory DBAs**

This event group is in the Inventory Increase (II) group.

##### **15.5.4.4.1 Feedwater Flow Increase All Pumps**

This event is designated as a CN-DBA event. The fault sequence name is Feedwater Flow Increase – All Pumps (CCF-FWFI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is II-CCF\_FWFI\_CCF-DL2\_CN-DBA.

### **Postulated Initiating Event**

One FW pump is normally operating and a second FW pump in standby. The RLC adjusts the pump speed to adjust FW flow to maintain RPV water level. This event assumes a spurious CCF that causes both FW pumps to increase flow to maximum speed that results in the maximum FW flow. Although not possible, the increase in flow is assumed to occur instantaneously.

### **Sequence of Events**

The fault sequence event summary:

- Both FW pumps increase to maximum flow causing RPV level increase
- RPC remains as-is at initial condition
- Level, pressure, and power increase
- Automatic FW isolation on high RPV level
- Scram on high simulated thermal power
- RPV pressure and level decrease
- Main steam isolation on low RPV pressure
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and remove decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated)
- Controlled state achieved

Table 15.5-18 lists the sequencing of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

The credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by increasing both feedwater pumps flow of to the maximum speed, resulting in the maximum FW flow. The initial conditions and total maximum flow for both FW pumps is provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated maximum FW pumps flow event is presented on Figures 15.5-76 through 15.5-81, and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response results. The increase in FW flow causes reactor level, pressure, and power to increase as reactor pressure control and reactor level control are unavailable. FW isolation occurs on high reactor level. Scram occurs on high simulated thermal power. Pressure decreases until MSRV isolation initiation occurs on low pressure. Pressure then increases and one ICS train is initiated on high pressure. Only one ICS train is needed to prevent RPV pressure from increasing and maintaining long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.4.2 Inadvertent Isolation Condenser Initiation – All Trains**

This event is designated as a CN-DBA event. The fault sequence name is Inadvertent Isolation Condenser Initiation – All Trains (CCF-DL4a-IIIC), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is II-CCF-IIIC\_CCF-DL2\_CN-DBA.

### **Postulated Initiating Event**

The ICS is normally in standby mode. This event assumes a spurious CCF that causes all ICS condensate return valves to open, resulting in introducing cold water into the reactor.

### **Sequence of Events**

The fault sequence event summary:

- All ICS condensate return valves open
- RLC and RPC fail as-is at the initial condition
- Cold ICS condensate water drains into the chimney
- Core flow, reactor pressure, and power decrease
- RPV water level increases due to RLC failing as-is
- FW isolation occurs on high RPV water level
- Scram occurs on low RPV pressure

- Main steam isolation occurs on low RPV pressure
- Controlled state achieved

Table 15.5-19 lists the sequence of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

The credited DL3 functions:

- DL3-23 – FW Isolation on High RPV Water Level
- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by opening all ICS condensate return valves in one second. The initial conditions are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The inadvertent initiation of all IC trains event is presented on Figures 15.5-82 through 15.5-87, and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response results. When the IC condensate return valves are opened, cold water is introduced into the chimney region, reducing reactor power, reactor pressure, and core flow. Reactor scram and MSRIV isolation initiation occurs on low pressure. After an initial reduction in reactor level, level begins to rise as reactor level control is unavailable. FW isolation occurs on high reactor level.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Sensitivity studies were performed at the ICS minimum initial temperature in Table 15.5-3 and for an assumed increase in ICS return line volume of 50%. There was no significant change in the event sequence or results. Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.5 Decrease in Reactor Coolant Inventory – DBAs**

This event group is in the Inventory Reduction (IR) group.

##### **15.5.4.5.1 Loss of Feedwater Flow**

This event is designated as a CN-DBA event. The fault sequence name is CCF Loss of FW Flow (CCF-LOFW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is IR-CCF-LOFW \_CCF-DL2\_CN-DBA.

#### **Postulated Initiating Event**

The event sequence assumes a spurious CCF causes the loss of all FW flow and a passive CCF of the DL2 function results in a fails as-is RPC.

#### **Sequence of Events**

The fault sequence event summary:

- Loss of FW flow causes RPV level and power to decrease
- RPV pressure decreases due to frozen RPC
- Scram occurs on low RPV level
- FW isolation on a loss of normal FW flow indication
- Main steam isolation on low RPV pressure
- All ICS trains initiate on low level
- Controlled state achieved

Table 15.5-20 lists the sequencing of events.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

#### **Systems Operation**

Credited DL3 functions:

- DL3-03 – Hydraulic Scram on Low RPV Level
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-14 – ICS Initiation on Low RPV Water Level
- DL3-39 – FW Isolation on Loss of Normal FW Flow

#### **Core and System Performance**

##### *Input Parameters and Initial Conditions*

The event is conservatively simulated by initiating a trip of all FW pumps. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The loss of FW flow is presented on Figures 15.5-88 through 15.5-93 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The trip of all FW pumps results in a reduction of vessel inventory, causing the pressure and vessel water level to drop. Reactor scram occurs on low RPV level. FW isolates on loss of normal FW flow. The MSRIV close on low pressure. RPV water level continues to decrease until ICS initiates. Three ICS trains are modeled to open. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train to start on low RPV level will not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs from metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

#### *Barrier Performance*

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

#### **15.5.4.5.2 Reactor Pressure Vessel Pressure Control Open**

This event is designated as a CN-DBA event. The fault sequence name is RPV Pressure Control Open (CCF-RPCO), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is IR-CCF-RPCO\_CCF-DL2\_CN-DBA.

#### **Postulated Initiating Event**

The event assumes all TCVs and TBVs are fully opened by a spurious RPC CCF. The event sequence assumes a passive CCF of the DL2 functions resulting in the RLC failing as-is.

#### **Sequence of Events**

The fault sequence event summary:

- All TCVs and TBVs open causing RPV pressure and power to decrease
- FW flow remains at 100% due to DL2 CCF
- Reactor scram and main steam isolation on low RPV pressure
- FW isolates on high RPV level

- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and removes decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated)
- Controlled state achieved

Table 15.5-21 lists the event sequence.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL3 functions:

- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by fully opening all TCVs and TBVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The simulated RPV Pressure Control Open event is presented on Figures 15.5-94 through 15.5-99 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The opening of the TCVs and TBVs results in a decrease in reactor pressure causing voids to increase and power to decrease. FW remains at 100% rated flow due to the RLC CCF. The reactor scrams and main steam isolates on low RPV pressure. RPV water level rises until FW isolates on high RPV water level. RPV pressure then rises due to decay heat. An ICS train initiates on high RPV pressure after the first IC train fails to actuate (assumed single failure). One ICS train sufficiently controls pressure.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequence associated with this event.

#### **15.5.4.6 Loss-of-Coolant Accidents – DBAs**

The scenarios for LOCA developed in Section 15.2.4 bound the CN-DBA sequences, demonstrating the fuel and containment integrity acceptance criteria are met for at least 72 hours using only passive heat removal systems.

The LOCA method used in containment analyses is described in NEDC-33922P BWRX-300 Containment Methods (Reference 15.5-2). The initial conditions and the modeling parameters are biased to account for uncertainties. The Defense Lines (DL) credited in the conservative LOCA breaks inside containment are identified in Table 15.5-49.

Meeting the acceptance criteria for fuel integrity is demonstrated by showing that level does not fall below the TAF, or fuel cladding temperature does not exceed the fuel cladding temperature during normal operating conditions.

As discussed in Section 15.2.4, all large breaks are isolated rapidly (10 seconds). Therefore, RPV inventory loss does not threaten fuel integrity in a large break LOCA. After RPV isolation, decay heat is removed by the isolation condensers directly from the RPV. The limiting parameters for large break LOCA events is containment pressure and temperature. Containment peak pressure reaches its peak value at approximately the time of RPV isolation.

The LOCA analyses demonstrate the core remains covered or fuel cladding temperature remains below the normal operating temperature for at least 72 hours using conservative assumptions for unisolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation.

#### **15.5.4.6.1 Main Steam Pipe Break Inside Containment, Conservative Case**

##### **Postulated Initiating Event**

A break in the main steam pipe occurs at an arbitrary location between the flow limiter, which is placed as close as possible to the RPV and the containment penetration. The most limiting break is the double-ended instantaneous guillotine break of the main steam pipe.

The break flow occurs from both ends of the break. Break flow from the RPV side of the break is choked at the flow limiter. The TSV/TCV is conservatively assumed to close rapidly because this results in retaining more energy in containment. Steam flows from the RPV to the closed TSV/TCV through the intact steam pipe and to the break location in the reverse direction to the normal flow through the broken steam pipe, discharging through the turbine side of the break. Steam flow from the RPV to the intact loop is also choked at the flow limiters. CIV closure is conservatively not credited, and the entire steam line volume inventory contributes to containment pressurization.

## Sequence of Events

The bounding scenario analyzed:

- Double-ended guillotine rupture of main steam line break inside containment concurrent with LOPP
- FW pump trip and coast down
- TSV or TCV start closing rapidly
- Scram initiated from high containment pressure
- Control rods start to insert
- ICS condensate return valve starts opening
- CUW stops
- Controls rods inserted sufficiently to diminish fission from prompt neutrons
- RIVs fully close
- Condensate return valve for one ICS train is fully open
- Peak containment pressure is reached
- Containment pressure starts decreasing

## Identification of Operator Actions

No operator action is required to mitigate the event.

## Systems Operation

As shown on Figure 15.5-102, break flow from the turbine side is generally higher than the break flow from the RPV side of the break because of the pipe inventory, and the break flow continues even after the RIVs are fully closed at 10 seconds.

Feedwater pumps are assumed to trip concurrent with the break initiation because the bounding scenario assumes LOPP concurrent with the pipe break. FW pumps are assumed to trip concurrent with the break initiation since the bounding scenario assumes LOPP concurrent with the break. One isolation condenser is started when the high containment pressure setpoint is reached, which occurs within 1 second. Although two isolation condensers are available, only one isolation condenser is credited to bound the large break case in the isolation condenser steam supply pipe. Figure 15.5-100 shows the heat removal rate of one IC exceeds the power generation decay heat after 20 seconds. As a result, reactor pressure decreases rapidly even after the break isolation shown on Figure 15.5-101. RPV level, labeled as "Collapsed Downcomer Level" on Figure 15.5-103 stabilizes well above the TAF in 3 hours. The decrease in the downcomer level during the first 3 hours is due to the gradual decrease in the void fraction in the core and chimney, not due to RPV water inventory losses. There is no RPV water inventory loss after the RIVs close. Fuel never heats up because the core remains covered throughout the event. After the fission power is diminished, fuel cladding temperature remains near saturation temperature.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Figure 15.5-104 shows the containment pressure in response to a large main steam pipe break inside containment. The break location is assumed away from the containment shell at the lowest main steam pipe elevation and directed upwards as discussed in LTR NEDC-33922P-A. This configuration maximizes containment pressure. The break location is assumed next to the containment shell and directed towards the containment shell in calculating the shell temperature on Figure 15.5-105 as discussed in LTR NEDC-33922P-A. This maximizes shell temperature.

The initial containment pressure includes a bias to account for uncertainties and is assumed at the containment high pressure setpoint for scram, reactor isolation and isolation condenser initiation. Although it appears that the setpoint is reached as soon as the break occurs, this is an artifact of the conservative initial condition assumption. A finite amount of time would have to elapse for containment pressure to reach the setpoint if the containment initial pressure is at the nominal pressure in normal operation. The pressure trend on Figure 15.5-104 shows that the containment pressure increases by 18.5 kPa in less than 1 second, indicating that the containment high pressure setpoint is reached in less than 1 second when the initial containment pressure is at the nominal pressure in normal operation. This confirms the break flow calculation assumption that containment high pressure setpoint will be reached in less than 1 second.

The initial reactor power in the conservative case calculations is 102% of the rated power to account for the power uncertainty. However, hot shutdown conditions may be more limiting for the mass release from the break because the initial RPV void fraction is lower resulting in higher liquid carryover to the break location. This also causes the break flow enthalpy to be lower. Both the rated initial conditions and hot shutdown initial conditions are analyzed. Figure 15.5-104 shows that the rated initial condition is more limiting for containment peak pressure.

PCCS does not require actuation, it is always in service and rejects heat to the equipment pool that is connected to the reactor cavity pool during normal operation. The calculations assume there is no heat loss from the containment shell to the concrete as discussed in the containment method LTR NEDC-33922P-A. Heat removal from containment atmosphere is by the containment shell heating up, by PCCS and through the containment dome to the pool. After RIV closure for large breaks, the only energy addition to the containment is due to the heat transfer from the RPV and hot piping walls. Because only one isolation condenser is sufficient to depressurize the RPV rapidly, heat load to containment also becomes small in the long term.

## **Core, System and Barrier Performance**

### *Results*

Event timing is summarized in Table 15.5-23. Key results are summarized in Table 15.7.3-3. The plots for RPV parameters and containment parameters are shown on Figures 15.5-100 through 15.5-105. The peak pressure is less than the design pressure with more than 20% margin. Containment shell temperature is also well below the containment shell design temperature of 166 °C. Peak accident pressure is approximately 322 kPaG (423 kPa) and half of the peak pressure is approximately 161 kPaG (262 kPa). As shown on Figure 15.5-104 **Error! Reference source not found.**, containment is depressurized rapidly, and containment pressure is reduced to 185 kPa at 6 hours. This meets the acceptance criterion for the containment response to pipe breaks that the containment pressure should be reduced to less than half of the peak pressure in 24 hours.

### *Barrier Performance*

There is no fuel damage as a result of an MSLB inside containment. The only activity available for release is normal reactor coolant concentration in the vessel and piping prior to the break.

### *Radiological Consequences*

The radiological consequences for a MSLB inside containment are bounded by the consequences for MSLB outside containment presented in Subsection 15.5.9.2.1.

#### **15.5.4.6.2 Feedwater Pipe Break Inside Containment, Conservative Case**

##### **Postulated Initiating Event**

A double-ended guillotine break occurs in the larger diameter segment of one feedwater pipe. This is more limiting than a break occurring in the smaller diameter FW pipe segments closer to the RPV. The bounding scenario is the same as that described in Subsection 15.5.4.6.1 for Main Steam Pipe Break Inside Containment

##### **Sequence of Events**

The bounding scenario analyzed:

- Double-ended guillotine rupture of the FW pipe break inside containment concurrent with LOPP
- FW trip and coast down
- TSV and TCV start closing rapidly
- Scram initiated from high containment pressure
- Control rods start inserting on scram initiation
- Condensate return valves on two ICS trains start opening
- Control rods are inserted sufficiently to diminish fission from prompt neutrons
- FWRIVs and CIVs are fully closed
- Peak containment pressure is reached
- IC valves are fully open

##### **Identification of Operator Actions**

No operator action is required to mitigate the event.

##### **System Operation**

RPV and containment response to a large FW break is similar to the RPV and containment response to a large main steam pipe break discussed in Subsection 15.5.4.6.1. Containment pressure reaches the containment high pressure setpoint for scram, reactor and containment isolation, and isolation condenser initiation in less than 1 second. Power generated by prompt fission is diminished in 3 seconds after the break. Condensate return valves in two of the three IC trains start opening in 1 second and fully open in 11 seconds. As shown on Figure 15.5-170, heat removal rate is much larger than the decay heat. As a result (shown on Figure 15.5-171), RPV pressure decreases much faster than the main steam pipe break case. Reactor water level is shown on Figure 15.5-172. The indicated water level stabilizes above the actual collapsed downcomer level. This is because the wide range level is off scale when the actual level falls below the lower tap and no longer indicates level. The actual collapsed downcomer level stabilizes well above TAF.

Break flow from the pump side decreases initially because the break location is far away from the pump and the enthalpy becomes saturated locally right after the break although the pump is still coasting down as shown on Figure 15.5-173. The break flow from the pump side recovers and exceeds that of the RPV break side. This is because the pipe water inventory and the pump coasting down. The break flow becomes zero when the CIV is closed at 10 seconds. Enthalpy value after this point is not meaningful because there is no break flow.

As in the main steam pipe break cases, containment pressure on Figure 15.5-174 and temperature on Figure 15.5-175 are calculated for a break location maximizing pressure and temperature.

An additional feedwater pipe break case was included accounting for the lower initial feedwater temperature because the containment peak pressure may be higher if the FW pipe break occurs when the plant is operating at reduced FW temperature. Break flow rate is higher at higher subcooling. However, break flow enthalpy is also lower. Figure 15.5-174 shows the containment pressure for normal FW temperature and reduced FW temperature. Normal FW temperature results in a higher containment pressure. The peak pressure for both cases is bounded by the peak pressure resulting from a main steam pipe break.

### **Core, System and Barrier Performance**

#### *Results*

Timing of events is summarized in Table 15.5-24. Plots for RPV and containment parameters are shown on Figures 15.5-170 through 15.5-175. Key results are summarized in Table 15.7.3-3 and show that the peak containment pressure and temperature resulting from feedwater pipe breaks are bounded by the main steam pipe breaks. Containment pressure and temperatures are less limiting than the main steam pipe cases and meet the acceptance criteria. Containment pressure calculated for FW pipe break at 6 hours is also less than half the peak containment pressure calculated for the main steam pipe case and decreasing.

#### *Barrier Performance*

There is no fuel damage as a result of an FWLB inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

#### *Radiological Consequences*

The radiological consequences for a FWLB inside containment are bounded by the consequences for FWLB outside containment presented in Subsection 15.5.9.2.2.

#### **15.5.4.6.3 Large Isolation Condenser Pipe Breaks Inside Containment**

An isolation condenser system break larger than the area of a 19 mm equivalent diameter pipe is detected by the leakage detection for each ICS train separately. When a break is detected in one ICS train, both the steam supply pipe and the condensate return pipe of the affected ICS train are closed. The stroke time and delay time assumed for the isolation condenser isolation valves in the analysis are the same as those for all other RIVs and bound all other equipment initiation delays starting from the time of the pipe break. The other two unaffected isolation condensers are available to remove decay heat. For conservatism, the analysis assumes only one of the two remaining isolation condensers is put in-service on high containment pressure. Therefore, the number of isolation condensers available in this case is only one, which is the same as the number of isolation condensers available in the main steam pipe break cases as analyzed for all breaks larger than a 19 mm diameter.

Although the isolation condenser steam supply pipe diameter may be as large as the MS flow limiter diameter, the break flow rate from an isolation condenser steam supply pipe break is less than the break flow rate from the main steam pipe break. This is due to the much larger inventory in the main steam piping connected to both ends of the break.

The liquid in the isolation condenser is subcooled and does not contribute to high energy discharge from the break. Therefore, the main steam pipe break for containment response is more limiting than the isolation condenser steam supply pipe break.

ICS condensate return pipe diameter is much smaller than the FW pipe diameter used in the FWLB analysis. Therefore, large breaks in the isolation condenser condensate return pipe are bounded by large breaks in the FW pipe or MS pipe.

Because the isolation condenser pipe breaks are bounded by either the MS pipe or the FW pipe breaks, no further analysis of isolation condenser pipe breaks is needed.

#### ***15.5.4.6.4 Small Steam and Liquid Pipe Breaks Inside Containment***

##### **Postulated Initiating Event**

A break area of  $\leq 19$  mm equivalent diameter remains unisolated. These breaks are analyzed for fuel integrity and containment integrity for at least 72 hours using conservative assumptions.

All liquid pipe break nozzles are at least 4 meters above TAF. A small pipe break on instrument lines may remain unisolated indefinitely.

A small liquid pipe break and a small steam pipe break have similar break flow rates after the level falls to 4 m above TAF. Because the isolation condensers depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat-up does not occur even without injection to the RPV. Containment heat removal occurs through the PCCS to the equipment pool and through the containment head to the reactor cavity pool.

##### **Sequence of Events – Small Steam and Liquid Pipe Breaks**

The bounding scenario analyzed:

- Small steam pipe break concurrent with LOPP
- Pressure controller remains as-is. TCV remains at initial position, turbine chest pressure decreases rapidly
- FW pump trips and coasts down
- Main steam pipe low pressure setpoint is reached
- Reactor scrams
- MSRIVs are fully closed
- Level decreases to Level 2
- Condensate return valves on two ICS trains are fully open
- Peak containment pressure is reached

##### **Identification of Operator Actions**

No operator action is required to mitigate the event.



## System Operation

The conservative cases assume LOPP concurrent with the pipe break, which is more limiting than the case where preferred power is available (discussed in Section 15.2). TCV/TSV closure is expected to occur as a consequence of the LOPP. However, TCV/TSV closure on LOPP is not credited. Rather, MSRV closure on low steam pipe pressure is conservatively credited in the analysis. The TCV/TSV back pressure is assumed to decrease rapidly maximizing the RPV water inventory loss to the turbine. Reactor scram also occurs on low steam line pressure accounting for the delays after the low steam pipe pressure is reached.

Isolation condenser condensate return valves start opening when the level falls to Level 2. As shown on Figure 15.5-115 and Figure 15.5-116, decay heat is removed by two isolation condensers. RPV depressurizes initially when the sum of the decay heat removal rate by the isolation condensers and the energy discharge from the break exceeds the decay heat power. Isolation condensers remove less power at lower pressure because of the lower temperature difference between the RPV steam and pool water. Reactor pressure stabilizes at a low value and the depressurization rate becomes very small. As shown on Figure 15.5-117, there is a rapid decrease in the collapsed downcomer level. This decrease is primarily due to the void collapse in the RPV. There are two small increases in level at approximately 69000 and 91000 seconds on Figure 15.5-117. These increases are due to void redistribution in the vessel. There is no increase in the RPV water inventory. As shown on Figure 15.5-117, downcomer collapsed level falls below TAF at 206000 seconds. However, the two-phase level in the core remains above TAF. Fuel remains wetted and thus never heats up.

The break mass and energy release are calculated assuming there is no back pressure. This assumption was made to bound breaks outside containment and accounts for the expected lower containment pressure than calculated because of the conservative assumptions used in the containment analyses. Even without break back pressure, Figure 15.5-119 shows that the break flow becomes very small in the long term.

Containment pressure calculated by using conservative assumptions and the small liquid pipe break flow without back pressure is shown on Figure 15.5-120. RPV pressure is also shown on the same figure. The calculated containment pressure increases to the RPV pressure at approximately 232800 seconds. Containment pressure is not higher than the RPV pressure because the break flow stops if the containment pressure becomes equal to the RPV pressure.

However, there is a potential that if ICS depressurizes the RPV faster than PCCS depressurizes containment in the absence of a break, reverse flow from containment to the RPV may occur. Non-condensables ingested into the RPV may collect in the isolation condensers and reduce their efficiency. Both the RPV and the containment could start repressurizing if back flow were to occur. In order to investigate this possibility, the containment pressure is calculated starting from 232800 seconds until the end of the 72 hour period assuming no break flow. The dashed line on Figure 15.5-118 shows that containment is rapidly depressurized in the absence of break flow. Because containment pressure decreases faster than the RPV pressure when there is no break flow, the RPV cannot depressurize below the containment pressure and reverse flow cannot occur. Energy released from the RPV through the break is a small fraction of the decay heat in the long term. A much larger fraction of the decay heat is removed by the isolation condensers. Therefore, RPV pressure calculated with and without a break are approximately the same at the time the RPV depressurizes to near containment pressure.

Figure 15.5-120 shows that containment pressure remains well below 262 kPa in the long term, which is 50% of the peak accident pressure calculated in Subsection 15.5.4.6.1.

Figure 15.5-108 shows that level remains above TAF for small steam pipe breaks. Containment response shown on Figure 15.5-110 and Figure 15.5-111 for a small steam pipe break is similar to the containment response in Figure 15.5-120 and Figure 15.5-121 for a small liquid pipe break.

A small pipe break on an isolation condenser steam pipe does not cause a more limiting core or containment response than an instrument line break. The small breaks conservatively credit only two of the three isolation condensers even though a break of  $\leq 19$  mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Insufficient steam in the RPV to feed the isolation condenser only occurs if the RPV is depressurized to the point where almost all of the steam escapes the break. This is the case if the RPV pressure is lower than that calculated for an instrument pipe break. However, in this case, the break flow is also less than the break flow calculated for an instrument pipe break, resulting in the isolation condenser small break less limiting than the instrument pipe break. Breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break. This is because the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser.

### **Core, System and Barrier Performance**

#### *Results*

For small steam pipe breaks, timing of events is summarized in Table 15.5-25. Key results are summarized in Table 15.7.3-3. The plots for RPV parameters and containment are shown on Figures 15.5-106 through 15.5-111.

All liquid pipe break nozzles are at least 4 meters above the TAF, and the timing of the events is summarized in Table 15.5-26. Key results are summarized in Table 15.7.3-3. The plots for RPV and containment parameters are shown on Figures 15.5-115 through 15.5-121. A small pipe break on the instrument lines may remain unisolated indefinitely. Since the isolation condensers depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat-up does not occur even without injection to the RPV.

#### *Barrier Performance*

There is no fuel damage as a result of a small liquid and feedwater breaks inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

#### *Radiological Consequences*

The radiological consequences for a small liquid or feedwater breaks inside containment are bounded by the consequences for small liquid and steam pipe breaks outside containment presented in Subsection 15.5.9.2.4.

### **15.5.5 Analysis of Design Extension Conditions Without Core Damage**

The bounding transient event selection in Subsection 15.2.3 determines the list of DEC events evaluated in the following subsections. Tables 15.5-3 through 15.5-4 provide input parameters and initial conditions used in the DEC analyses. Table 15.5-5 provides the DLs used in the DEC analyses. The analysis is performed consistent with the DSA analysis approach described in Subsections 15.1.3 and 15.2.4.

#### **15.5.5.1 Control Rod Drop Accident – Practically Eliminated**

The BWRX-300 GNF-2 fuel has a core design that is similar to the BWR operating fleet.

The FMCRD uses a bayonet style coupling that requires a 45-degree rotation to uncouple. Since the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to uncouple from the FMCRD during reactor operation. The hollow piston is the component within the FMCRD coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual FMCRD separation switches that sense that the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur.

Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the likelihood of rapid rod withdrawal. The ball check valve is a SC1 DL3 function because it prevents:

- Reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line
- The loss of pressure from the underside of the hollow piston
- The generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion

Normal rod movement and the rod withdrawal rate are limited by the FMCRD. The rod control system controls rod patterns and provides control rod blocks limiting the rate and amount of reactivity addition for control rod movement.

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows limited reactivity additions from rod withdrawal. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented using a bayonet style coupling, CRD mechanism latches, and CRD separation switches. As a result, the CRDA and control rod ejection event have been practically eliminated.

#### **15.5.5.2 Pressure Increase – DECs**

##### **15.5.5.2.1 Closure of One Main Steam Reactor Isolation Valve**

This event is designated as an EX-DEC event. The fault sequence name is Closure of One MSRIV (1MSRIVC), and the Fault Sequence ID is PI-1MSRIVC\_CCF-Hydraulic-Scram\_EX-DEC.

#### **Postulated Initiating Event**

The PIE is the same as the BL-AOO event. The analysis assumes a CCF hydraulic scram failure. The control rods enter the core using the CRDM run-in function. This event demonstrates that the CRDM run-in function performs the FSF control of reactivity without hydraulic scram.

## Sequence of Events

The fault sequence event summary:

- One MSRIV closes causing pressure and power to increase
- Hydraulic scram signal on MSRIV position. Scram fails
- Hydraulic scram on any signal fails
- MSRIV in the second steam line closes on leak detection indication (assumed because it makes the event more severe)
- CRDM run-in initiation on high flux after scram signal
- All ICS trains initiate on high flux after scram signal
- FW pumps trip on high flux after scram signal
- Controlled state achieved

Table 15.5-27 lists the sequencing of events.

## Identification of Operator Actions

No operator action is required to mitigate the event.

## Systems Operation

Credited DL2 functions:

- DL2-21 – Anticipatory Hydraulic Scram Signal on MSRIV/MSRIV Position (scram fails)
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Credited DL4a functions:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

## Core and System Performance

### *Input Parameters and Initial Conditions*

The event is simulated by initiating closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated closure of one MSRIV is presented on Figures 15.5-124 through 15.5-129 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. The pressure increase is limited due to the initiation of ICS, FW pump trip and CRDM run-in that occur on high flux after scram signal (i.e., indications of high power level post scram initiation). The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The

acceptance criteria are provided in Section 15.3. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

This event resulted in the highest peak cladding temperature and peak vessel pressure. Sensitivities are performed to examine cliff edge effects. Sensitivities on key initial conditions and phenomena that impact cladding temperature and peak vessel pressure described in the TRACG Application (Reference 15.5-3) are applied separately by at least one standard deviation:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under
- Critical quality used in boiling transition correlation
- Hot rod power

Results indicate no significant cliff edge effects. No excessive vessel pressure and no core damage occurs.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

#### **15.5.5.2.2 Complex Sequence of Generator Load Rejection or Turbine Trip**

This event is designated as an EX-DEC event. The fault sequence name is Complex Sequence Generator Load Rejection or Turbine Trip (LR-TT) plus CCF-Mechanical CRD, and the Fault Sequence ID is CSS-LR-TT\_CCF-Mechanical-CRD\_EX-DEC. This event demonstrates that with multiple failures to insert independent control rods with diverse motive forces combined with a very frequent PIE, that the remaining control rods perform the FSF reactivity control.

#### **Postulated Initiating Event**

The postulated initiating event is the same as for the LR-TT AOO event. Additionally, the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram. No other failures are assumed.

#### **Sequence of Events**

The fault sequence event summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- Anticipatory scram occurs on indication of a turbine trip or load rejection signal, but half of the control rods fail to insert
- RPC opens TBVs to control pressure

- RLC maintains level
- Controlled state achieved

Table 15.5-28 lists the sequence of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by initiating a generator load rejection or turbine trip that results in a fast closure of the TCVs and/or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The simulated generator load rejection / turbine trip is presented on Figures 15.5-130 through 15.5-135 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for reactor pressure response. Automatic anticipatory reactor scram occurs following indication of a generator load rejection or turbine trip with half of the rods failing to insert. The neutron flux increases rapidly because of the void reduction caused by the pressure increase.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

A controlled state is achieved with steam bypassed to the main condenser and fed back into the reactor by feedwater. This is maintained for a significant amount of time as long as power is available. Given the conditions, operators initiate additional CRDM run-in signals or manually insert CRDM to insert the remaining control rods into the core. If operator actions are unsuccessful, operators inject boron to shut the reactor down. Another option available to the operators is to decrease power by reducing FW flow. With reduced FW flow, reactor water level decreases, reducing core flow and reducing reactivity through void reactivity feedback until the steam flow matches the FW flow.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

#### **15.5.5.2.3 Loss of Condenser Vacuum with CCF Hydraulic Scram**

This event is designated as an EX-DEC event. The fault sequence name Loss of Condenser Vacuum (LOCV), CCF-Hydraulic-Scram and the Fault Sequence ID is PI-LOCV\_CCF-Hydraulic-Scram\_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

### **Postulated Initiating Event**

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are assumed.

### **Sequence of Events**

The fault sequence event summary:

- Loss of vacuum results in a turbine trip
- TSVs close quickly causing pressure and power increase
- Hydraulic scram signal on either turbine trip or high main condenser pressure scram fails
- Hydraulic scram fails on any signal
- CRDM run-in initiation occurs on high flux after scram signal
- RPC opens TBVs to control pressure
- RLC maintains level
- ICS initiates on high flux after scram signal
- FW pumps trip on high flux after scram signal
- TBVs close on high main condenser pressure
- Controlled state achieved

Table 15.5-29 lists the sequencing of events.

### **Identification of Operator Actions**

No operator action is required to mitigate the event.

### **Systems Operation**

Credited Defense Line functions:

DL2:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand (scram fails)

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The loss of condenser vacuum results in a turbine trip. The event is simulated by initiating a turbine trip that results in a fast closure of the TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### **Results**

The simulated loss of condenser vacuum is presented on Figures 15.5-136 through 15.5-141 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is initially limited by the TBVs opening. The pressure increase is limited due to the initiation of ICS, FW trip, and CRDM run-in that occur on high flux after scram signal.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).



#### **15.5.5.2.4 Loss-of-Preferred Power with CCF Hydraulic Scram**

This event is designated as an EX-DEC event. The fault sequence name is Loss-of-Preferred Power (LOPP), CCF-Hydraulic-Scram and the Fault Sequence ID is PI-LOPP\_CCF-Hydraulic-Scram\_EX-DEC. This event demonstrates that the CRDM run-in function that performs the FSF controlling reactivity without the hydraulic scram.

#### **Postulated Initiating Event**

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are postulated in the event.

#### **Sequence of Events**

The fault sequence event summary:

- LOPP results in generator output breakers opening
- TCVs close quickly
- Feedwater pumps lose power
- Hydraulic scram signal fails on either generator load rejection or low electric bus voltage
- Hydraulic scram fails on any signal
- CRDM run-in initiation on high flux after scram signal
- TBVs close on loss of power
- ICS initiates on high flux after scram signal
- Controlled state achieved

Table 15.5-30 lists the sequencing of events.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

#### **Systems Operation**

Credited Defense Line functions:

DL2:

- DL2-01 – Maintain Target Pressure
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand (scram fails)
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

## **Core and System Performance**

### *Input Parameters and Initial Conditions*

The Loss-of-Preferred Power results in the generator output breakers opening and a loss of power to the feedwater pumps. The event is simulated by initiating a FW pump trip and a load rejection that results in a fast closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

### *Results*

The simulated Loss-of-Preferred Power with CCF hydraulic scram failure is shown on Figures 15.5-142 through 15.5-147 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The rapid closure of the TCVs results in a pressure increase. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is initially limited by the opening of the TBVs. The TBVs later close, ICS initiates, and CRDMs run-in on high flux after scram signal. The ICS continues to limit the pressure increase.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

## **15.5.5.3 Reactivity and Power Distribution Anomalies – DEC**

### **15.5.5.3.1 All Control Rod Withdrawal at Power (ACRW)**

This event is designated as an EX-DEC event. The fault sequence name is CCF - All Control Rod Withdrawal at Power- All Rods (CCF-ACRW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is RI-CCF-ACRW\_CCF-DL2\_EX-DEC.

### **Postulated Initiating Event**

All control rods inserted in the core start to withdraw due to rod control spurious CCF. A passive CCF of DL2 technology platform results in DL2 function failure.

## Sequence of Events

The fault sequence event summary:

- All control rods start to withdraw resulting in a power increase
- ATLM (Automatic Thermal Limiting Monitor) and MRBM (Multi-Channel Rod Block Monitor) fail to block rod withdrawal
- RPC and RLC fail as-is at the initial condition
- Scram on STP power or neutron flux
- After scram, the immediate challenge to cladding and RCPB integrity is over
- RPV pressure decreases
- RPV level decreases temporarily due to the void collapse in the core and chimney
- Sensed level increases due to continuing FW flow and flashing in the downcomer
- FW isolation occurs on high RPV level
- Main steam isolation occurs on low RPV pressure and pressure slowly increases
- One ICS train initiates on high pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-31 lists the sequencing of events.

## Identification of Operator Actions

No operator action is required to mitigate the event.

## Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV pressure
- DL3-11 – ICS Train 1 initiation on High RPV pressure

## Core and System Performance

### *Input Parameters and Initial Conditions*

The event is simulated by withdrawing all the control rods in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

### *Results*

The simulated ACRW is presented on Figure 15.5-148 through 15.5-153. The analysis results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response result.

When the control rods are withdrawn, the power increases and scram occurs on high simulated thermal power or neutron flux. The RPV water level increases and FW is isolated. RPV pressure decreases and the MSRIVs close. Eventually the RPV pressure increases, and one ICS train initiates. The pressure remains well within the DBA event RCPB pressure acceptance criteria in Subsection 15.3.1. There is no challenge to containment.

The long-term core cooling capability is assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed and there is no core damage.

### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

#### **15.5.5.3.2 Inadvertent Single Control Rod Withdrawal at Power (ICRW) - DEC**

This event is designated as an EX-DEC event. The fault sequence name is Inadvertent Control Rod Withdrawal at Power - single rod (ICRW), and the Fault Sequence ID is RI-ICRW\_CCF\_DL2\_EX-DEC.

### **Postulated Initiating Event**

A control rod inserted in the core is withdrawn due to a failure. A passive CCF of DL2 results in failure of the DL2 functions.

### **Sequence of Events**

The fault sequence event summary:

- Single rod (with highest reactivity worth) starts to withdraw
- ATLM and MRBM fail to block the rod withdrawal
- RPC and RLC are assumed to function normally because this prolongs the event and makes it more severe for cladding temperature effects
- Reactor power increases but the scram level is not reached
- Local power and cladding temperature increase
- Power and the cladding temperature reach a stable level
- Operator action to initiate scram is expected due to the high power level (Not credited in analysis simulation as these DL functions demonstrate achieving and maintaining a controlled state)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- After the scram, no credit is taken for RPC or RLC (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- RPV pressure decreases and the RPV level increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- FW isolation occurs on high RPV level (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- Main steam isolation occurs on low RPV pressure and pressure slowly increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- One ICS train initiates on high pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-32 lists the sequencing of events.

#### **Identification of Operator Actions**

No operator action is credited.

#### **Systems Operation**

Credited DL3 functions:

- DL3-17 – MSRV/MSIV isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-11 – ICS Train 1 Initiation on High RPV Pressure

#### **Core and System Performance**

##### *Input Parameters and Initial Conditions*

The event is simulated by withdrawing a single control rod in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR and the hot rod power are consistent with the reference core design.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

##### *Results*

The simulated ICRW event is presented on Figures 15.5-154 through 15.5-159. The analysis results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response result.

When the control rod is withdrawn, the power increases. The RPV level and pressure vary insignificantly because RLC and RPC function to maintain level and pressure preventing scram, thus maximizing the impact of fuel cladding temperature increase.

In this event, the PCT values are well below the DBA acceptance criteria. The long-term core cooling capability is assured because the effects are local. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria.

Sensitivities are performed examining cliff edge effects. Key initial conditions are conservatively biased to cause transition boiling even though this does not occur at the nominal / realistic conditions associated with DEC conditions. With initial CPR conservatively biased low (by approximately 0.05), and the hot rod power (LHGR) conservatively biased high (approximately 30%), local high cladding temperatures occur in hot rods in a few high power bundles located near the control rod withdrawn in error. This result is expected in fuel cladding failure in a very limited number of rods. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

#### *Barrier Performance*

There is no challenge to the RCPB and containment. The fuel cladding may experience local failures if initial LHGR and CPR are more severe. The predicted number of rod failures is limited to high-powered fuel rods in a few high-powered bundles near the control rod withdrawn in error. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

#### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

#### **15.5.5.4 Decrease in Reactor Coolant Inventory – DEC**

This event is in the IR group and is designated as an EX-DEC event. The fault sequence name is FW Isolation (CCF-FWI-DL3) and the Fault Sequence ID is IR-CCF-FWI-DL3\_EX-DEC.

#### **Postulated Initiating Event**

The event sequence assumes a spurious CCF isolates all FW flow and a passive CCF of the DL3 functions.

#### **Sequence of Events**

The fault sequence event summary:

- FW flow isolation causes RPV water level and power to decrease
- RPC maintains RPV pressure
- Scram and MSRV isolation on sustained low FW flow
- ICS pressure control initiates on high RPV pressure
- Controlled state achieved

Table 15.5-33 lists the event sequence.

#### **Identification of Operator Actions**

No operator action is required to mitigate the event.

#### **Systems Operation**

Credited Defense Line functions:

DL2:

- DL2-01 – Maintains Target Pressure
- DL2-42 – Anticipatory Hydraulic Scram on Sustained Low FW Flow
- DL2-31 – ICS Pressure Control on High Reactor Pressure

DL4:

- DL4a -12 – MSRIV/MS CIV Isolation on Sustained Low FW Flow

### **Core and System Performance**

#### *Input Parameters and Initial Conditions*

The event is simulated by initiating a conservatively fast isolation of all FW flow. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

#### *Results*

The CCF FW isolation event is presented on Figures 15.5-160 through 15.5-165 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The loss of FW flow results in a reduction of vessel inventory, causing the power and RPV water level to decrease. RPC maintains reactor pressure. Reactor scram and main steam isolation occurs based on sustained low FW flow. ICS pressure control initiates based on high RPV pressure.

The long-term core cooling capability is conservatively assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is conservatively assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs from metal water reaction.

#### *Barrier Performance*

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

#### *Radiological Consequences*

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

### **15.5.6 Analysis of Design Extension Conditions with Core Damage**

The analysis of DEC with core damage are addressed in the Level 2 PSA described in Section 15.6.4.

### **15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool**

The analysis fault sequences associated with the fuel pool are DEC analyzed in the Level 1 PSA described in Subsection 15.6.3.

### 15.5.8 Analysis of Fuel Handling Events

The Fuel Handling Accident (FHA) is categorized as a non-reactor group DEC event. The fault sequence ID is FHA\_EX-DEC. The event occurs as a result of a failure of the fuel assembly lifting mechanism, resulting in the drop of a raised irradiated fuel assembly onto the reactor core or into the fuel pool. The dropped irradiated fuel assembly results in cladding failure in the dropped and impacted bundles. The sequence of events for the postulated FHA is provided in Table 15.5-34.

The dose results are conservatively compared to the criterion in Table 15.7.6-1 to demonstrate compliance to the DBA acceptance criterion.

#### **Fuel Damage**

Because of the complex nature of the impact and the resulting damage to fuel assembly components, predicting the number of failed rods is not possible. For this reason, a simplified energy approach is taken. Numerous conservative assumptions are made to assure that the number of failed rods is conservatively analyzed.

The key assumption for the FHA is that during a refueling operation, a fuel assembly is moved over the top of the reactor core or fuel pool. While the fuel grapple is in the over-hoist condition with the bottom of the assembly at the maximum height allowed when using the fuel handling equipment, the main hoist cable and a redundant cable fails. This results in the fuel assembly, the fuel grapple mast and head falling on top of the core impacting a group of four fuel assemblies. The grapple head and mast are fixed vertically to the dropped assembly so that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly might bend. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel or fuel pool.

During refueling operations, the bounding radiological event is a drop over the core due to the maximum drop height.

Fuel rod failure is assumed at 1% circumferential strain. The associated axial strain is  $(0.01)/\nu$ , where  $\nu$  is Poisson's ratio, plastic deformation is assigned a value of 0.5, and the energy per rod failure is expressed by:

$$E_f = \sigma_y \times \epsilon \times V \quad (\text{Equation } E_f)$$

Where:

$E_f$  = energy per rod failure

$\sigma_y$  = yield stress

$\epsilon$  = axial cladding strain

$V$  = volume of fuel cladding

Kinetic energy is calculated for the dropped fuel bundle that accounts for the influences of buoyancy and resistance from the reactor cavity pool water. Finite element analysis simulations are used to determine the kinetic energy based on the drop distance in air or water. The simulation results revealed that the drop distance of a fuel bundle in air is greater than 2.3 m (7.5 ft), while the kinetic energy of the fuel bundle drop in water is less than 50% of that in air. When the bundle drop height is 10.4 m (34.0 ft) the energy is approximately 22% of that in air. This analysis credits a 50% reduction in the kinetic energy of the dropped bundle although the limiting case drop height correlates to a larger reduction.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The fuel assembly wet weight and the mast wet weight are used in applying the 50% kinetic energy reduction to the fuel assembly due to dropping through water that is expressed as:

$$E_1 = \frac{h_{\text{drop}} \times (W_{\text{fuel}} + W_{\text{mast}})}{2}$$

Where:

$E_1$  = energy from initial drop

$W_{\text{fuel}}$  = weight of fuel bundle

$W_{\text{mast}}$  = weight of refueling mast

$h_{\text{drop}}$  = drop height

Substituting numerical values yields Equation  $E_1$ :

It is assumed that half of the energy is absorbed by the cladding. The ratio of the cladding to the non-fuel mass for GNF2 fuel is 0.4997. The calculated yield strength using the methodology described above (see Equation  $E_1$ ) is 37.503 kgf-m/rod (271.26 ft-lbf/rod). Therefore, the number of failed rods in the impacted assemblies from the initial drop is 31 rods.

Additional energy is generated in a secondary impact as the bundle falls over from a vertical orientation to a horizontal orientation, and damages additional rods in the impacted bundles. Incorporating the 50% reduction due to the kinetic energy in water is expressed as:

$$E_2 = 50\% \times (h_{\text{fuel}} \times W_{\text{mast}} + \frac{1}{2} \times h_{\text{fuel}} \times W_{\text{fuel}})$$

Where:

$E_2$  = energy of dropped bundle and mast from secondary impact

$h_{\text{fuel}}$  = height of refueling mast

Fifty percent of the kinetic energy is absorbed by the impacted assemblies resulting in the number of failed rods in the impacted assemblies of 5 rods.

All the rods in the dropped assembly are assumed to fail and are full-length. GNF2 fuel assemblies contain both full-length and part-length rods accounting for the difference in lengths resulting in 85.6 effective full-length rods per bundle. The number of failed rods used in determining the radiological consequences is 128 failed fuel rods or 1.495 failed fuel bundles.

There are 240 fuel bundles in the BWRX-300 core. The fraction of the core damaged in an FHA is determined by:

$$\text{Core Damage Fraction} = \frac{1.495 \text{ bundles damaged}}{240 \text{ bundles in the core}} = 6.23\text{E-}03$$

### **Core Inventory of Isotopes**

A BWRX-300 core inventory of isotopes is calculated using the Oak Ridge National Laboratory code ORIGEN2, version 2.1 and the BWRUE.LIB cross section library in units of Ci/MWth for the bulk operating parameters in Table 15.5-35.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

A subset of the more than 700 isotopes from this inventory are used to model DBA dose consequences. The 60 isotopes used for DBAs are the dominant contributors to immersion and inhalation doses from airborne activity released during a DBA. This set of nuclides consist of 54 isotopes identified in WASH-1400 (NUREG-75/014) and 6 isotopes identified in SAND-85-2575 (NUREG/CR-4467). This is the group of isotopes typically used for alternate source term (AST) dose evaluations.

The BWRX-300 reactor is subcritical for at least 24 hours prior to initiating refueling operations. The BWRX-300 core inventory of the 60 dose-significant isotopes after 24 hours of decay are shown in Table 15.5-36.

### ***Gap Fractions***

For events that are non-LOCAs where some fuel damage is postulated like the FHA, the fractions of the core inventory assumed in the fuel rod gap region for the various radionuclides are taken from USNRC Regulatory Guide (RG) 1.183 and reported in Table 15.5-37.

### **Radial Peaking Factor**

The radioactive material available for release in an FHA is assumed in the analysis to come from assemblies with peak inventory. To simulate this assumption, the inventory is scaled up by the maximum power Radial Peaking Factor (RPF). This represents the maximum achievable operational power history immediately preceding shutdown. Based on prior experience with GNF2 cores, a representative RPF value of 1.70 is assumed for this analysis.

### ***Activity Released from the Fuel***

All particulate isotopes are retained by the water in the fuel pool or reactor cavity water. Thus, only the noble gases and the gaseous form of iodine are available to escape the water. The activity released from the fuel shown in Table 15.5-38. The release is the mathematical product of the core power ( $E_f$ ), RPF, gap fractions (Table 15.5-37), and core damage fraction. With this information, the release from the fuel is calculated as shown in Table 15.5-38.

### ***Pool Scrubbing (Decontamination)***

Credit is taken for retention of some of the released fission gas in the water where it is released from the damaged fuel rods. Because the depth of water above reactor cavity pool and SFP is greater than 7.01 m (23.0 ft), the simple Decontamination Factor (DF) model from USNRC RG 1.183, Appendix B is used.

The activity released from the surface of the reactor cavity pool, shown in Table 15.5-39, is calculated by applying the RG 1.183 DFs to the activity released from the fuel in Table 15.5-38.

### ***Transport in the Reactor Building***

The radioactivity released from the reactor cavity pool is assumed to mix instantaneously with the free air volume of the refueling outage floor and crane area, which is the intermediate volume between the FHA release from the water and the environment. The refueling outage floor and crane area has an approximate air volume of 9,620.25 m<sup>3</sup> (339,736 ft<sup>3</sup>) taking into account the equipment footprint.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The analysis conservatively ignores mixing to account for the possibility of inadequate mixing over the short term. Therefore, crediting for mixing is reduced by 50% by reducing the volume of airspace. There is no mechanical means to ensure the refueling outage floor and crane area airspace is well mixed. To account for the 50% volume reduction of airspace:

$$\text{Credited Refueling Area Volume} = (50\%) \times 9620.25 \text{ m}^3 = 4,810.13 \text{ m}^3 (169,868 \text{ ft}^3)$$

This doubles the concentration of contamination in the leakage from the RB to the environment.

### ***Release Assumptions***

To implement this in the model, a single leakage rate from the reactivity cavity or fuel pool ensures the entire release is transported to the environment from the RB in two hours.

To simulate the leakage to the environment, the following equation is used to calculate a corresponding leakage rate.

This rate is calculated using the equation below, and setting to time (variable "t") to 120 minutes as follows:

$$\frac{C(t)}{C_{ss}} = e^{-Qt/V}$$

Where:

$C(t)$  = evacuation volume transported to the environment at time t

$C_{ss}$  = evacuation volume at time t=0

$C(t)/C_{ss}$  = fraction of evacuation volume remaining at time t (unitless)

$Q$  = constant flow rate out of space ( $\text{m}^3/\text{s}$ )

$V$  = volume of air transported to the environment ( $\text{m}^3$ )

$t$  = time duration of dilution period in minutes (minutes)

As  $C(t)$  approaches steady state or complete transport of the evacuation volume (assume a value of 0.1 %),  $C(t)/C_{ss}$  approaches a value of zero and the previous equation can be simplified to:

$$0.001 = e^{-Qt/V}$$

Substituting the known values and solving for Q:

$$-\ln(0.001) \times \frac{V}{t} = Q$$

The flow rate that transports 99.9% of the contamination released to the RB refueling outage floor and crane area can be transported to the environment in two hours at a flow rate of  $267.9 \text{ m}^3/\text{min}$  ( $9778.4 \text{ ft}^3/\text{min}$ ).

### ***Dispersion in the Environment***

Transport to the EAB and Automatic Action Zone (AAZ) from the BWRX-300 is simulated using atmospheric dispersion factors ( $\chi/Q$  values) that are established for the Darlington site. Dispersion and deposition is determined in the ADDAM code based on 5 years of DNNP site-specific meteorological data. The meteorological data set used complies with the requirements of CSA N288.2:19. The DBA  $\chi/Q$ s are set equal for the OPG 350 m Emergency Zone (EZ).

### ***Dose Calculation***

Because the inventory of isotopes released from the surface of the reactor cavity pool is already determined in Table 15.5-39. The release from the pool to the RB in Table 15.5-39 is modeled at time  $t=0$ .

### ***Breathing Rates***

The postulated FHA breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

### ***Decay and Daughtering***

This analysis assumes a decay time of 24 hours prior to the removal of spent fuel during refueling, and credit for this decay period is taken in the initial core inventory Table 15.5-36.

Decay and daughtering of nuclides during the FHA are assumed in the ADDAM model for the duration of the event (30 days).

### ***Dose Conversion Factors***

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. Consistent with REGDOC-2.4.1, the dose contributions include:

- External radiation from cloud and ground deposits
- Inhaled radioactive materials
- Skin absorption of tritium

The ADDAM code uses dose conversion factors that comply with CSA N288.2.

### ***Results***

The BWRX-300 FHA, which is classified as a DEC, meets the Darlington site DBA dose acceptance criteria with considerable margin for the most critical group as demonstrated in Table 15.7.6-1.

## **15.5.9 Analysis of Radioactive Releases from a Subsystem or a Component**

### **15.5.9.1 Analysis of Liquid and Gaseous Radioactive Waste System Scenario Releases**

The radioactive liquid tank and offgas system (OGS) failures are addressed as process system failures. The analysis for these releases is described in Chapter 11, Subsections 11.2.9 and 11.3.13 and Table 11.3-4.

#### ***Liquid Tank Failure***

To prevent tank leakage from exiting the Radwaste Building (RWB), the collection and sample tanks are enclosed with a concrete wall barrier. If a tank failure occurs, the concrete wall containment area prevents any liquid radioactive waste from exiting the RWB. The contents of the tank drain to the nearest radioactive drain system, and any venting is collected in the buildings ventilation exhaust system and processed accordingly.

#### ***Offgas System Failure***

The limiting BWRX-300 OGS failure is an inadvertent bypass of the charcoal delay beds. An evaluation was performed, and the resulting public dose consequences are well below the REGDOC-2.5.2, Section 4.2.1 limits as shown in Chapter 11, Table 11.3-4.

#### **15.5.9.2 Analysis of Loss-of-Coolant Accidents Outside Containment**

As discussed in Subsection 15.5.4.6, the scram and RPV isolation trips occur for the large breaks outside containment within the same time as breaks inside containment.

For large breaks, timing of the break detection is less than 1 s for breaks outside containment that is the same as the time to reach the drywell high pressure setpoint. Because reactor scram and isolation valve closures for breaks inside containment also occur for breaks outside containment, the event progression is no different for breaks outside containment than inside containment. For main steam pipe breaks, the break flow rate calculated for breaks inside containment is also used for breaks outside containment because the MSCIV closure is not credited in the main steam pipe break inside containment. For FW pipe breaks, the only difference between the pipe breaks inside and outside of containment is the closure of the FWCIV. For a FW pipe break outside containment, break flow includes flow from the reactor as well as the pipe inventory.

The isolation condenser pipe break outside containment is limited to the flow passing through the orifices in the steam distribution pipes. During normal operation prior to the break, condensate return valves are closed and remain closed. Isolation steam supply pipe has a guard pipe outside containment so that break flow in the supply pipe upstream of the orifice is not discharged outside containment. A break in the isolation condenser is followed by a discharge of the subcooled water in the isolation condenser into the pool, followed by steam break flow passing through the orifice in the steam supply pipe until the isolation condenser RIVs close on break detection.

Since the small break analyses inside the containment do not credit containment back pressure, the mass and energy release calculated for breaks inside the containment are bounding for breaks outside the containment.

##### **15.5.9.2.1 Main Steam Line Break Outside Containment**

The thermal-hydraulic response of RPV and containment for a MSLB outside containment is bounded by the response to a break inside containment as presented in Subsection 15.5.4.6.1.

##### *Event Description*

The MSLB accident is assumed to occur in the steam tunnel which is the interface between the RB and Turbine Building (TB). The postulated event assumes that a main steam line instantaneously and circumferentially breaks downstream of the outermost MSIV. The plant is designed to immediately detect such an occurrence, initiate isolation of all main steam lines including the broken line. The release of reactor steam and water from the break is assumed to blowdown into the TB airspace where it is released to the environment instantaneously as a ground-level release, with no building holdup credited.

##### *Source Term*

There is no fuel damage as a result of an MSLB outside containment. The only activity available for release from the break is that which is present in the reactor coolant prior to the break. Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design, etc.) are determined based on ANSI/ANS-18.1.

Two cases are considered for conditions when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the release to the environment is assumed instantaneous and without holdup from the TB coolant radiation concentrations.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

For postulated accidents the design basis BWRX-300 reactor coolant concentrations are adjusted to account for two conditions that may exist when the accident begins:

1. The concentration that is the maximum equilibrium value (typically 0.2  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted under plant Operating Limits and Conditions (OLCs).
2. The concentration that is the maximum iodine spike value (typically 4.0  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted under plant OLCs.

The plots for RPV are shown on Figures 15.5-100 through 15.5-103 and for containment parameters Figures 15.5-104 through 15.5-105.

#### *Radiological Consequences*

The dose consequences are calculated for the MSLB accident at the proposed BWRX-300 exclusion zone.

The dose consequences of the MSLB are calculated using the ADDAM computer code (refer to Section 15.5.1.2).

#### *Mass Release*

The total mass of coolant released is the amount in the line and connecting lines at the time of the break plus the amount that passes through the RPV isolation valves and the outboard containment isolation valve prior to closure. The masses of coolant and steam released from the postulated MSLB based on preliminary thermal-hydraulic analysis are:

Liquid Release = 15,878 kg

Steam Release = 2,400 kg

Table 15.5-44 provides the activity release from this coolant volume.

This analysis conservatively assumes that all of the liquid released from the break flashes to steam and is available for transport to the environment along with the steam released from the break. The mass release duration from the MSLB is equal to the maximum closure time of the containment isolation valve of 10 seconds.

#### *No Holdup Release to the Environment Flow Rate*

No holdup release to the environment is assumed with a total of 99.9% transport of the TB airspace over a period of 10 minutes after the event. *Breathing Rate*

The postulated MSLB breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

#### *Decay and Daughtering Nuclides*

Decay and daughtering of nuclides during the MSLB are credited in the dose model for the duration of the event.

#### *Dose Conversion Factors*

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

#### *Results*

This radiological consequence is calculated for the 30-day whole-body dose at the exclusion zone. The results are listed in Table 15.7.7-1 for comparison to the radiological acceptance criteria.

The MSLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-1 for comparison against the radiological acceptance criteria.

#### **15.5.9.2.2 Large Feedwater Pipe Break Outside Containment**

Flow from the RPV side of the break is bounded by feedwater breaks inside containment because of the longer pipe length. The higher pressure losses occur for a break outside containment. No operator actions are required to mitigate the event.

##### *Event Description*

The FWLB accident occurs in the BWRX-300 steam tunnel that interfaces between the RB and TB. An instantaneous circumferential break of a feedwater line is postulated. The plant is designed to immediately detect such an occurrence and initiate FW line isolation. The energetic release of reactor water from the break is assumed to blowdown into the TB airspace where it is instantaneously released to the environment as a ground-level release, with no building holdup credited.

##### *Source Term*

There is no fuel damage as a result of an FWLB outside containment in the BWRX-300. The only activity available for release from the break is that which is present in the reactor coolant prior to the break.

Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design, etc.) are determined based on ANSI/ANS-18.1.

For DBAs, the design basis BWRX-300 reactor coolant concentrations are adjusted to account for two conditions that may exist when the accident begins:

1. The maximum equilibrium value concentration (typically 0.2  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted under plant OLCs for continued full power operation.
2. The maximum Dose Equivalent I-131 concentration (typically 4.0  $\mu\text{Ci/gm}$ ) permitted under plant OLCs and corresponds to the conditions of an assumed pre-accident spike.

##### *Mass and Energy Release.*

The total mass of coolant released is the amount in the line and connecting lines at the time of the break plus the amount that passes through the RPV isolation valves and the outboard containment isolation valve prior to closure. The masses of coolant and steam released from the postulated FWLB based on preliminary thermal-hydraulic analysis are:

Liquid Release = 18,090 kg

Steam Release = 733 kg

Break Flow Enthalpy = 1289.1 kJ/kg

The mass release vs. time is shown on Figure 15.5-167.

##### *Flashing of Reactor Coolant to Steam*

The FW break fluid enthalpy varies at saturation conditions at differing pressures. As the coolant exits the break into the steam tunnel, some fraction of the coolant released from the break flashes to steam and becomes available for release to the environment.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The flash fraction (FF) is determined using a constant enthalpy process:

$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$

Where:

$h_{f1}$  = The enthalpy of fluid within the FW line at normal operating conditions in kJ/kg

$h_{f2}$  = The specific enthalpy of water at atmospheric pressure in kJ/kg

$h_{fg}$  = The heat of vaporization at 100 °C in kJ/kg

The flashing fraction applied to the liquid release is 0.390.

*No Holdup Release to the Environment Flow Rate*

No holdup release to the environment is assumed with a total of 99.9% transport of the TB airspace over a period of 10 minute after the event.

*Breathing Rates*

The postulated FWLB the breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

*Decay and Daughtering Nuclides*

Decay and daughtering of nuclides during the FWLB are credited in the dose model for the duration of the event.

*Dose Conversion Factors*

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

*Radiological Consequences*

The dose consequences were calculated for the FWLB accident outside containment at the proposed BWRX-300 exclusion zone.

Two cases are considered for conditions that may exist when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the release to the environment is instantaneous and without TB holdup.

Radiation concentrations in BWRX-300 reactor coolant and steam are determined based on ANSI/ANS-18.1.

The dose consequences of the FWLB are calculated using the ADDAM computer code (refer to Section 15.5.1.2).

This radiological consequence is calculated for the 30-day whole-body dose at the exclusion zone. The results are listed in Table 15.7.7-2 for comparison to the radiological acceptance criteria.

*Results*

The FWLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-2.



#### **15.5.9.2.3 Shutdown Cooling System Pipe Break Outside Containment**

The shutdown cooling system (SDC) (Chapter 9, Subsection 9A.2.3) provides decay heat removal for refueling or maintenance. The SDC provides decay heat removal at normal and lower reactor operating pressures. SDC is subjected to high energy conditions for a short time (less than 2% of the plant operating conditions). The system piping is assigned a medium energy line due to the short time that it is subjected to high energy conditions.

The SDC connects to the feedwater system between the feedwater containment isolation and feedwater isolation control valve outside containment. Due to the smaller SDC piping diameter, the feedwater pipe break outside containment discussed previously in Subsection 15.5.9.2.2 bounds the SDC pipe break outside containment.

#### **15.5.9.2.4 Large Isolation Condenser Pipe Breaks Outside Containment**

The isolation condenser pipe configuration outside the containment is under development. For the worst-case scenario, the large isolation condenser pipe break outside containment may be larger than the large isolation pipe breaks inside containment. The mass and energy release from the isolation condenser pipe breaks outside containment are still bounded by the isolation condenser pipe breaks inside containment. Breaks inside containment remain bounding because the isolation condenser pipe breaks do not utilize any DL3 functions that depend on containment parameters, and the containment back pressure is not credited in any of the isolation condenser pipe breaks inside containment.

As is the case for isolation condenser pipe breaks inside containment, core response is not a concern since the break is isolated rapidly. The consequences of large isolation condenser pipe breaks outside containment require an evaluation for the loads, pressures, and temperatures outside containment, and radiological consequences resulting from normal operation coolant activity.

#### **15.5.9.2.5 Small Breaks Outside Containment**

Since the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment.

However, dose analyses are performed for breaks outside containment for two line break cases:

- ICS line break
- Instrument line break

### **ICS Line Break Dose Consequences**

#### *Event Description*

The ICS line break accident analyzed for offsite dose consequences is a postulated break of an ICS steam supply pipe in the ICS pool on the operating deck of the RB. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line within 5 seconds, and fully isolate the break in 10 seconds. Blowdown steam from the break is directed to the two heat exchangers in one ICS unit and the liquid mass in the heat exchangers is expelled to the ICS pool where it mixes with the pool water without flashing. The energetic release of reactor steam from the break is assumed released to the environment instantaneously as a ground-level release, without holdup.

### *Source Term*

There is no fuel damage as a result of an ICS line break outside containment. The only activity available for release from the break is that which is present in the reactor coolant prior to the break.

Radiation concentrations in reactor coolant and steam are adequate for use in design basis calculations (such as shielding, equipment design, etc.) and are determined based on ANSI/ANS-18.1.

For postulated accidents the design basis reactor coolant concentrations are adjusted accounting for two conditions that may exist when the accident begins:

- The concentration that is the maximum equilibrium value (typically 0.2  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted under plant OLC.
- The concentration that is the maximum iodine spike value (typically 4.0  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted under plant OLC.

### *Mass Release*

The mass of steam released from the postulated ICS line break is 400.8 kg and is based on preliminary GEH thermal-hydraulic analysis.

### *Release Duration*

The mass release duration from the ICS line break is equal to the maximum closure time of the isolation valves that is 10 seconds.

The mass of steam released from the break is transported to the environment over a 10 minute period which adequately simulates instantaneous transport.

### *Breathing Rates*

The postulated ICS line break breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

### *Decay and Daughtering Nuclides*

Decay and daughtering of nuclides during the ICS line break are credited in the dose model for the duration of the event.

### *Dose Conversion Factors*

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

### *Results*

The ICSLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-3.

### **Instrument Line Break Dose Consequences**

The dose consequences of an Instrument Line Break Accident (ILBA) at the exclusion zone considers two cases that may exist when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the line break release to the environment from the RB is instantaneous and without holdup.

The dose consequences of the ILBA are calculated using the ADDAM computer code.

### *Event Description*

A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment in the RB. The ILBA analysis assumes that the event cannot be isolated, and no fuel damage occurs. The resulting activity is released to the environment directly from the RB with no credit for holdup or filtration. Primary coolant flows at the maximum rate for a typical instrument line that has a 1/4" flow restricting orifice. Saturated water flows in the instrument line into containment that flashes to steam, resulting in the maximum iodine release.

### *Source Term*

There is no fuel damage as a result of this accident. The only activity released from the break is that present in the reactor coolant prior to the break.

Radiation concentrations are determined using ANSI/ANS-18.1-2020 for the reactor coolant and steam used in design basis calculations for shielding and equipment design.

The design basis reactor coolant and steam concentrations are adjusted to account for two conditions that may exist when the postulated accident occurs:

1. The maximum equilibrium value concentration (typically 0.2  $\mu\text{Ci/gm}$  Dose Equivalent I-131) permitted by the plant OLCs.
2. The maximum iodine spike value (typically 4.0  $\mu\text{Ci/gm}$  Dose Equivalent I-131) concentration permitted by the plant OLCs.

The ILBA Airborne Release Source Term for equilibrium and iodine spike is provided in Table 15.5-43A and 15.5-43B, respectively. Only the equilibrium concentrations of iodine and iodine spike activity releases differ. The concentrations of the remaining isotopes are not impacted by iodine behavior.

### *Coolant Mass Release*

The mass of steam released 408.0 kg from the postulated ICS line break is used for the ILBA based on preliminary GEH thermal-hydraulic analysis. The release duration is equal to the maximum closure time of the isolation valves in the ICS break outside containment that is 10 seconds. The mass of released break steam is transported to the environment over a 10-minute period that adequately simulates instantaneous transport.

### *Release Duration*

The steam mass released is transported from the RB to the environment over a 10-minute period that adequately simulates instantaneous transport with no holdup. As a result, the steam release duration to the environment follows the coolant mass release duration.

### *No Holdup Release to the Environment Flow Rate*

No holdup release total transport of 99.9% from the TB airspace to the environment occurs over a period of 10 minute after the event.

### *Breathing Rates*

The postulated ILB breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

### *Decay and Daughtering*

Decay and daughtering of nuclides during the MSLB event are credited in the dose model for the duration of ILBA event.

### *Dose Conversion Factors*

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

### *Results*

The BWRX-300 ILB dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-4.

#### **15.5.9.3 BWRX-300 Out of Core Criticality Analysis**

A representative Out Of Core Criticality (OCC) accident scenario has been analyzed in the BWRX-300 Out of Core Criticality Safety Analysis Demonstration that shows that the dose consequences at the site boundary do not exceed the generic criteria to trigger a public evacuation (see Canadian Guidelines for Intervention During a Nuclear Emergency - Reference 15.5-12). Criticality analyses assess fuel handling activities outside the core for the GE High Density Fuel Storage Racks (HDFS) and the RAJ-II Inner Container (IC). The HDFS rack assessment for the storage of GNF2 fuel analyses resulted in a storage rack maximum  $k_{\text{eff}}$  ( $k_{\text{max}}(95/95)$ ) less than the USL of 0.95 for normal and credible abnormal operation. The OCC analysis demonstrates compliance to REGDOC-2.4.3, Section 2.3.2.2, Part 3 (Reference 15.5-13) by computationally investigating a representative accident scenario that complies with REGDOC-2.4.3, Section 16.4. The representative OCC accident scenario is postulated to occur inside the RB on the refueling floor due to the unsafe stacking of RAJ-II-ICs.

### **Method of Analysis**

The estimated total number of fissions follows the use of a simplified models approach described in BSI ISO 16117:2013 (Reference 15.5-14):

- The estimate of the number of fissions should be based on simplified options providing “order of magnitude” values
- This estimate should rely on the collective experiences from past criticality accidents (Annex B) and criticality experiment results (Annex C) and the possible use of simplified formulae (Annex D)
- When a simplified model is used, the consistency of its area of applicability with the chosen assumptions of the postulated criticality accident shall be justified and documented

The parameters extracted from ISO 16117:2013, Table B.2 (Reference 15.5-14) for the purpose of this analysis is provided in Table 15.5-45. The total number of fissions allows for determining the neutron and photon source term magnitudes that are used subsequently in developing MCNP-06P models that are used to assess the dose consequence.

### **Computational Models**

Computational models have been developed in MCNP-06P to assess the neutron and photon dose consequence of the postulated OCC accident scenario. MCNP-06P is the GEH controlled version of the Los Alamos National Laboratory code MCNP6 (Reference 15.5-15). MCNP6 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. All models employ ENDF/B-VII.0 cross section libraries. Compliance with CSA standards is demonstrated and confirmed.

The first computational model is developed for the purpose of tracking the particle flux/dose as a function of distance in air from the accident location (see Figure 15.5-168). The second computational model is developed to study particle flux attenuation as a function of concrete thickness present to predict shielding behavior in the refueling floor area where the accident occurs. The relative decrease in neutron/photon radiation dose rates is utilized to approximate the dose consequence as a result of shielding present.

### **Source Term**

The total number of fissions shown in Table 15.5-46 is used to determine the neutron and photon source term magnitudes utilized in the MCNP-06P calculations. The energy distribution of the neutron source is governed by the Watt energy spectrum. The photon source term is discretized into two energy groups with even probability. The first group is the average energy emitted per prompt fission photon, and the second group is the maximum prompt fission photon energy.

### **Radiation Dose Rates**

For all computational models, the neutron and photon dose rates are computed by utilizing the ANSI/ANS-6.1.1-1977 (Reference 15.5-11) particle flux to dose conversion coefficients in the F4 tallies to yield units of mSv/H. The time-integrated dose is calculated by scaling the total dose rates by the accident duration to yield time-integrated dose units of mSv. The resultant doses at the site boundary are then directly compared to the generic criteria from Canadian Guidelines for Intervention During a Nuclear Emergency (see Table 15.5-47) to show that no protective actions are necessary to ensure the safety of the public in the event of the postulated representative OOC accident.

### **Results**

The results for the total integrated dose provided in Table 15.5-47 are compared directly to the generic criteria for implementing actions to protect the public provided in Table 15.5-48. The total integrated dose is plotted on Figure 15.5-169 with respective error bands to highlight dose behavior as a function of distance from the OOC accident location, with and without the attenuation provided by concrete shielding.

The total integrated doses provided in Table 15.5-46 was conservatively based upon the total number of fissions actually occur during the accident excursion timeframe. These results are deemed to be conservative based on the following:

1. The unsafe stack of RAJ-II ICs is originally confined in space by the gravitational, normal, and static friction forces. The fission force magnitude that occurs during the accident within the critical system will overwhelm the gravitational and normal and frictional forces holding the bundles in their original locations ultimately displacing the fuel far enough apart to yield a subcritical system.
2. There are established RAJ-II IC stacking limits and fuel handling patterns in place to avoid unsafe ICs stacking. In order for an OOC scenario to occur, workers would violate operating procedures or have inadequate training.

In the event that workers violate established operational procedures, there are natural and physical barriers in place mitigate the likelihood of an OOC event sequence. The following physical barriers are DL1 design measures that prevent an OOC occurrence:

1. The time it takes to obtain and arrange RAJ-II ICs in an 8x8 stack on the refuel floor is significant. The ICs must each be handled via crane and cannot easily be displaced. It is quite unlikely that other workers in the reactor building would ignore the workers unsafely stacking ICs during this entire duration. There are safety checks in place and verifications made during the process that prohibit unsafe stacking.
2. The new fuel receipt and handling plan schedule is provided to the plant that contains the number of fuel bundles that arrive on site for the first 14 years of plant operation. There is a limit to how many ICs are allowed on the refuel floor at one time.
3. The criticality analysis was based on an average bundle fuel enrichment of 5 wt% U-235 for each RAJ-II IC in the stack. This is conservative and unlikely as the average enrichment for the BWRX-300 core is less than 5 wt% U-235.

## **Conclusions**

The dose consequence of a representative criticality accident scenario in accordance with REGDOC-2.4.3, Section 16.4.1 shows that the dose consequence at the site boundary does not exceed the generic criterion that would trigger a public evacuation in accordance with Canadian Guidelines for Intervention During a Nuclear Emergency (Reference 15.5-12).

### **15.5.10 Analysis of Internal and External Hazards**

The internal and external hazards portion of the PSA (refer to Subsection 15.6.2.4) explicitly analyzes radionuclide release accidents initiated during power and shutdown operation for internal and external hazards. Malevolent acts are addressed in the Security Annex.

The internal and external hazards is presented in Chapter 3. Sections 3.4 and 3.3, respectively. The design features to mitigate these hazards are provided for the specific hazards in the following subsections.

#### **15.5.10.1 Analysis of Internal Hazards**

The description of internal hazards and design features are described in the following sections:

Fires – Chapter 3, Section 3.4.1 hazard identification, design features Chapter 9A, Section 9A.6

Explosions – Chapter 3, Section 3.4.1 hazard identification, design features in Chapter 9B, Section 9B

Toxic Gas – Chapter 3, Subsection 3.4.1 hazard identification, MCR and SCR design features in Chapter 6, Section 6.4

Internal Flooding – Chapter 3, Subsection 3.4.2 hazard identification, structural design in Chapter 9B, Section 9B

Internal Missiles – Chapter 3, Subsection 3.4.3 hazard identification, structural design in Chapter 9, Section 9B, equipment and pipe design features in Chapter 3, Section 3.6, and Chapter 5

High Energy Line Breaks (HELB) – Chapter 3, Subsection 3.4.4 hazard identification, HELB evaluation in Chapter 3, Subsection 3.6.7

#### **15.5.10.2 Analysis of Natural External Hazards**

The description of external hazards and the structural design features are described in the following sections:

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Seismic – Chapter 3, Subsection 3.3.1 hazard identification, structural design in Chapter 9, Section 9B

Extreme Weather (Hurricanes, Tornadoes, Wind, etc.) – Chapter 3, Subsection 3.3.2 hazard identification, structural design in Section 9B

Hydrogeological (Floods) – Chapter 3, Subsection 3.3.3 hazard identification, structural design in Chapter 9, Section 9B

Aircraft Impact – Chapter 3, Subsection 3.3.4 hazard identification, structural design in Section 9B

Missiles – Chapter 3, Subsection 3.3.5 hazard identification, structural design in Section 9B

Fires – Chapter 3, Subsection 3.3.6 hazard identification, design features in Chapter 9, Section 9A.6

Explosions – Chapter 3, Subsection 3.3.6 hazard identification, structural design in Chapter 9, Section 9B

Toxic Gas – Chapter 3, Subsection 3.3.6 hazard identification, design features in Chapter 6, Section 6.4

#### **15.5.10.3 Analysis of External Human-Induced Hazards**

The external human-induced hazards is described in Chapter 3, Subsection 3.3.7. Malevolent acts is addressed in the PSAR Security Annex.

#### **15.5.11 References**

- 15.5-1 NEDE-32177, "TRACG Qualification," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-2 NEDO-33922-A, "BWRX-300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-3 NEDC-33987P, "TRACG Application for BWRX-300," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-4 GOTHIC Thermal Analysis Package Technical Manual, Version 8.3(QA), November 2018.
- 15.5-5 GOTHIC Thermal Analysis Package Qualification Report, Version 8.3(QA), November 2018.
- 15.5-6 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 15.5-7 NEDC-32082P SH 0001, "BWR Steady State Thermal Hydraulic Methodology (ISCOR)," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-8 CANDU Owner's Group, Inc., SQAD-20-5065, "ADDAM 1.4.2 User's Manual," April 2021.
- 15.5-9 CSA N288.2-19, "Guidelines for calculating the radiological consequences to the public of a release of airborne radioactive material for nuclear reactor accidents," CSA Group
- 15.5-10 ANSI/ANS-18.1-2020, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society.
- 15.5-11 ANSI/ANS-6.1.1-1977, "American National Standard: neutron and gamma-ray flux-to-dose-rate factors," American National Standard.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- 15.5-12 Government of Canada, "Canadian Guidelines for Intervention During a Nuclear Emergency," Minister of Health.
- 15.5-13 CNSC Regulatory Document REGDOC-2.4.3, "Nuclear Criticality Safety."
- 15.5-14 BS ISO 16117, "Nuclear criticality safety - Estimation of the number of fissions of a postulated criticality accident," British Standards Institution.
- 15.5-15 LA-UR-17-29981, "MCNP User's Manual<sup>(R)</sup> - Code Version 6.2," Los Alamos National Security, LLC.
- 15.5-16 NEDC-33939P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," GE-Hitachi Nuclear Energy Americas, LLC.



## **15.6 Probabilistic Safety Assessment**

A principal element of the design process is the development and results of the Probabilistic Safety Assessment (PSA). The PSA provides an integrated review of the plant design, operational safety, and complements the results of the deterministic analyses. The PSA measures how the safety of plant design and operation prevents the risk of releasing radionuclides to the environment. The PSA supports risk-informed design development and with the DSA, demonstrates the success of the design in achieving the design objectives. The PSA assesses design vulnerabilities and optimizes the design using a graded approach consistent with CNSC REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants," (Reference 15.6-1).

CNSC REGDOC-2.4.2, Requirement 3.2 (Reference 15.6-1) requires a Level 1 and Level 2 PSA. The BWRX-300 PSA meets the performance requirements of ANSI/ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment For Advanced Non-Light Water Reactor Nuclear Power Plants Publication," (Reference 15.6-2).

Other important sources of PSA guidance are in USNRC Regulatory Guide (RG) 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 15.6-3), USNRC RG 1.206, "Applications for New Power Plants" (Reference 15.6-4), International Atomic Energy Agency (IAEA) SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants" (Reference 15.6-5), and IAEA SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Plants," (Reference 15.6-6) provide guidance for performance of Level 1 and Level 2 PSAs. Canadian Standards Association (CSA) N290.17-17, "Probabilistic safety assessment for nuclear power plants" (Reference 15.6-7) provides high-level guidance and guidelines for performing a PSA. These PSA regulatory guides align with CNSC REGDOC-2.4.2 (Reference 15.6-1). The PSA level of detail expands as the design details are finalized in compliance with CNSC REGDOC-2.4.2, Requirement 3.8 (Reference 15.6-1) that states that the level of detail must be consistent with testing, maintenance, configuration management programs, and the intended PSA uses.

CNSC REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility," Section 4.4.5, (Reference 15.6-8) requires that "the PSAR includes a deterministic safety analysis, a PSA and a hazards analysis that demonstrate all levels of D-in-D" are addressed and confirms that the facility's design is capable of meeting applicable dose acceptance criteria and safety goals.

CNSC REGDOC-1.1.2 (Reference 15.6-8) and CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 15.6-9), require performance of Hazard Analysis, PSA, and DSA. Section 15.6 is a summary of the work performed for the Hazard Analysis Screening and PSA. The PSA is updated for the operating licence application.

The PSA is updated as additional design and site-specific information becomes available for the operating licence application.

### **15.6.1 General Approach to Probabilistic Safety Assessment**

The PSA risk assessments are divided into two levels:

- Level 1 calculates Core Damage Frequency (CDF) inside containment
- Level 2 calculates Small Release Frequency (SRF) and Large Release Frequency (LRF), with the associated release magnitude

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The Level 1 PSA consists of:

1. Initiating Events (IE) –events that may challenge fuel cooling. The initiating events may occur when the unit is at-power, low power, or shutdown.
2. Event Tree Analysis – plant response analysis for mitigating systems credited for maintaining fuel cooling. Mitigating systems failures result in fuel cooling failures and subsequent core damage.
3. Fault Tree Analysis – models mitigating system failure. Fault tree analysis requires human reliability inputs and component failure including CCF.
4. Accident Sequence Quantification – determines frequency of the core damage end states.
5. Uncertainty and Sensitivity Analyses – assesses the statistical and phenomenological uncertainties. Sensitivity analyses determines initiating events, mitigating systems, or phenomena that has major impact on the results.

Level 2 PSA consists of:

1. Reactor Coolant System (RCS) / Containment Response Analysis
2. Developing an interface between Level 1 and Level 2 PSA
3. Identifying and modeling safety functions and operator actions
4. Performing containment performance analysis
5. Developing Containment Event Tree
6. Phenomena Analysis – consisting of accident progression analysis
7. Source term analysis
8. Quantifying the model and results for LRF and SRF and consequences
9. Performing uncertainty and sensitivity analysis for SA release category, character, and quantification.

The PSA is divided into internal and external events. The internal events occur within the plant (e.g., loss-of-coolant, loss of feedwater, internal fire, etc.). External events occur from outside the plant (e.g., seismic, high wind, flood etc.).

Fuel Pool events address spent fuel damage.

The PSAR PSA scope includes:

A. Level 1 PSA hazards:

1. Internal Events are events at-power, low power and shutdown, internal flood at-power, and internal fire at-power
2. External Events are seismic event at-power, and high wind at-power

The development of BWRX-300 PSA models are conducted with Electric Power Research Institute (EPRI) Integrated Risk Technology suite of software packaged under the Phoenix Architecture, which includes the following:

- CAFTA for event tree, fault tree and cut set development and viewing purposes
- Phoenix Application Programming Interface (API) for model integration and one-top development
- PRAQuant for quantification processes, as well as for sensitivity runs

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The PSA model is integrated and quantified using the computer codes CAFTA, PRAQuant, and Fault Tree Reliability Evaluation eXpert (FTREX). These computer codes have been demonstrated throughout the industry to produce appropriate results. No method-specific limitations have been identified with regard to the software tools or the methodology implemented to quantify the model.

Using CAFTA, the BWRX-300 PSA model is developed by merging all model event trees, system fault trees, IEs, passive feature failure results, and associated basic event databases. Using PRAQuant, a top logic fault tree is created. All system fault trees are merged with the top logic fault tree. The model is quantified to arrive at event sequence frequencies for the various accident sequence end states and release categories.

Figure 15.6-1 demonstrates the principal steps in PSA model development.

The BWRX-300 PSA methodology approaches each PSA element. The level of detail for some technical areas does not include all steps but may refer to specific guidance documents or reports. For example, in the fire PSA, the description refers to USNRC NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1 "Fire Probabilistic Risk Assessment Methods Enhancements" (Reference 15.6-27) that provides details of each fire PSA step. Not all the described methods are applied early in the design.

Hazard and scenarios that are estimated as lower risk for the final design may be initially analyzed early in the design with simplified scoping evaluations, with a detailed PSA performed later in the design phase or during the site-specific PSA. Where these simplifications are applied, this is generally noted in the document. The DNNP PSA evolves with the design and is subject to IAEA-TECDOC 1106, "Living Probabilistic Safety Assessment (LPSA)" (Reference 15.6-10). The PSA is maintained and updated as the design matures and procedures are developed, when the plant(s) is built, Operating Experience (OPEX) is gathered and applied for any design or operating modifications.

External hazards analysis addresses possible combinations of external hazards. For example, high wind causing loss of the external grid is considered in the high wind analysis. Other site-specific combinations (such as snowstorm blocking the diesel generator combustion air intake and frazil ice blocking sea water cooling) are assessed.

The design phase PSA covers both at-power (full and reduced power levels) and low power and shutdown operations, which include outages and transitions between each plant operating state (POS) in accordance with the guidance of CNSC REGDOC-2.4.2, Section 3.10 (Reference 15.6-1). The design phase PSA includes main systems and components (including the reactor and turbine I&C logic), main operator actions, and the relevant event and system dependencies, interconnections, and CCF relationships.

The development of DNNP PSA models is conducted with the EPRI Integrated Risk Technology Phoenix Architecture. The computer codes CAFTA and PRAQuant, and FTREX are used to integrate and quantify results into PSA model. These computer codes are known throughout the industry to produce appropriate results. No method-specific limitations have been identified regarding the software tools or the methodology implemented to quantify the model. Chapter 3, Appendix 3G provides additional discussion about all the computer programs used in the PSA.

The BWRX-300 PSA model is developed by merging all model event trees, system fault trees, initiating events, passive feature failure results, and associated basic event databases. A top logic fault tree is created. All system fault trees are merged with the top logic fault tree. The model is then quantified to arrive at event sequence frequencies for the various accident sequence end states and release categories.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

At a minimum, the following areas are addressed in the design phase PSA:

- DNNP risk results compared to Safety Goals:
  - CDF
  - SRF
  - LRF
- Initiating event selection, grouping, and frequency for each analyzed plant operating state
- Event sequence modeling and quantification for accident sequence end state and release category
- System analysis and system models
- Success criteria
- Component reliability data (including CCF data and modeling principles)
- Human Reliability Analysis (HRA)
- Uncertainty and sensitivity analysis
- Containment analysis
- Mechanistic source term analysis for each release category
- Model documentation

The design phase PSA includes the PSA models as well as documentation describing plant characteristics important to safety, modeling assumptions and techniques, model structure, data values sources used in the model, and the analysis results are explained making it possible to review and replicate the analyses.

CNSC REGDOC-2.4.2, Section 3.3 (Reference 15.6-1) requires that the PSA be conducted under the management system or quality assurance program established in the licensing basis. This section shows how PSA output (in concert with the DSA) provides input into the design that may affect the PSA models. All work for the design phase PSA is performed in accordance with the requirements of the BWRX-300 Quality Assurance Program Plan.

### **Regulatory PSA Requirements**

A PSA is required to be conducted in accordance with the defined requirements for a licence to construct or operate a reactor facility per CNSC REGDOC-2.4.2, Section 1.1 (Reference 15.6-1). CNSC REGDOC-1.1.2, Section 4.4.1 (Reference 15.6-8) states that the PSA is required to support the PSAR.

In reviewing CNSC REGDOC-2.4.2 (Reference 15.6-1) PSA objectives, six of the eight objectives are being met. The two outstanding objectives require detailed Emergency Operating Procedures (EOPs) and Severe Accident Management Guideline (SAMG) procedures that are developed during the final safety analysis. Per CNSC REGDOC-2.4.2, Section 2.0 (Reference 15.6-1), the PSA objectives are:

- A. Providing a systematic analysis in order to give confidence that the reactor facility's design will align with the fundamental safety objectives as established in IAEA No. N-SF-1, Fundamental Safety Principles, including to protect people and the environment from harmful effects of ionizing radiation.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- B. Demonstrating that a balanced design has been achieved; this can be demonstrated as achieved if no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk.
- C. Providing confidence that small changes of conditions that may lead to a catastrophic increase in the severity of consequences (cliff-edge effects) are prevented.
- D. Providing assessments of the quantitative safety goals (the probabilities of occurrence for severe core damage states, and the assessments of the risks of radioactive releases to the environment) as defined in CNSC REGDOC-2.5.2 (Reference 15.6-9) or as established in the licensing basis for the facility.
- E. Providing site-specific assessments of the probabilities of occurrence and the consequences of external hazards.
- F. Identifying facility vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents or mitigate their consequences.
- G. Assessing the adequacy of EOPs throughout the nuclear power plant lifetime.
- H. Providing insights into the severe accident management program used in the development, implementation, training, and optimization of accident management strategies and measures.

This section provides the hazard analysis summary presenting the methodology, hazard identification and the screening results in compliance with CNSC REGDOC-1.1.2, Section 4.4.1 (Reference 15.6-8).

### **Regulatory Safety Goals**

The CNSC defined safety goals for core damage, small release, and large release frequencies are defined in CNSC REGDOC-2.5.2 (Reference 15.6-9), as follows:

#### *Core Damage Frequency*

The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than  $10^{-5}$  per reactor year.

#### *Small Release Frequency*

The sum of frequencies of all event sequences that can lead to a release to the environment of more than  $10^{15}$  becquerels of iodine-131 shall be less than  $10^{-5}$  per reactor year. A greater release may require temporary evacuation of the local population.

#### *Large Release Frequency*

The sum of frequencies of all event sequences that can lead to a release to the environment of more than  $10^{14}$  becquerels of cesium-137 shall be less than  $10^{-6}$  per reactor year. A greater release may require long-term relocation of the local population.

The DNNP PSA is performed iteratively with the design development to evaluate and improve the risk aspects of the design.

The PSA and severe accident evaluations specific objectives demonstrate that the BWRX-300 is designed with state-of-the-art safety features, incorporating highly reliable and available passive safety functions with significant redundancy and diversity meeting all established safety goals with margin.

The DNNP design PSA uses information available from the BWRX-300 plant design and procedures that meets the intent of CNSC REGDOC-2.4.2, Section 3.4 (Reference 15.6-1), for the PSA to reflect the as-built and as-operated plant. Periodic updates of the PSA (CNSC REGDOC-2.4.2, Section 3.5 (Reference 15.6-1)) are required after the plant begins operation and every five years thereafter. Component failure data and IE frequencies are based on generic U.S. industry data compared to the BWRX-300 design. As required by CNSC REGDOC-2.4.2, Section 3.7 {Reference 15.6-1}, this data and the assumptions used in developing the PSA are selected realistic as practical. Because site or programmatic information is not available during the design, the PSA analyses contains conservative elements (e.g., human error probabilities, maintenance delay or postponement, component failure rates, flood and fire initiation, propagation, and effects).

#### **15.6.1.1 Principal Steps in the Probabilistic Safety Assessment Model Development**

A design process that is “risk-informed” incorporates the results of risk evaluations and PSA where risk is evaluated for both the frequency and consequences of a possible event. In the nuclear industry, where design, operation, and maintenance are generally conducted using a prescriptive set of requirements, risk evaluations provide information on potential vulnerabilities of the design that may not be revealed by the deterministic design process. In using risk evaluations to assess a design, the risk results add to or supplement the design resulting in an increase in its overall reliability and safety. Using the PSA during the design process ensures that no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk. The application of risk information and PSA to supplement and improve a design during the design process is to risk-inform the design.

A design process that is “performance-based” relies on a process or equipment measurable outcomes as evidence of meeting a requirement or objective. One advantage of a performance-based approach to design is the flexibility available to meet the outcome requirement or objective. This contrasts with a prescriptive requirement where the adherence to the requirement infers an acceptable outcome. A process that is both risk-informed and performance-based uses both the process to inform and improve the design and the design process.

The full benefits of risk-informed, performance-based methods applied to the design for a new reactor are achieved through a disciplined and quantitative evaluation of options considered during the design process. The effectiveness of risk-informed, performance-based methods rely on the presence of four process elements that ensure a minimum necessary structure is available for these evaluations. These four process elements are:

- Design criteria or requirements
- Systems engineering process
- D-in-D
- Sequence-based risk assessment (using PSA)

The design process uses the PSA results at each stage in the design to confirm the design adequacy and to evaluate alternatives being considered during the system engineering and design process. The PSA interfaces with the design requirements, the system engineering process, and D-in-D evaluations. The PSA process and use is described throughout this section.

The internal events at-power model is developed first and covers these principal steps. The internal and external hazard models and the low-power and shutdown models are developed based on the internal events at-power model files.

By using the ASME/ANS standards for LWR PSA, the DNNP PSA is also required to follow what is normally referred to as a Level 2 assessment, which accounts for the response of containment

following core damage events. IAEA SSG-4 (Reference 15.6-6) provides guidance on assessing safety for low power and shutdown states. Where applicable, these guides are used to inform the DNNP PSA efforts.

#### **15.6.1.2 Internal and External Events and Level 1 PSA**

The internal and external hazards portion of the PSA analyzes radionuclide release accidents initiated during power and shutdown operation for the following hazards:

- Internal Hazards:
  - Internal floods
  - Internal fires
- External Hazards:
  - High winds
  - Seismic events
- Other hazards (internal and external). Risk assessment is based on the results of screening analyses that identifies potential significant contributors.

The internal and external hazard analyses are initially bounding assessments that show significant design margin for these hazards. The frequencies of internal events are based on generic industry data or bounding site data and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the internal and external hazards analyses to the extent possible, using logic flags or additional fault tree modeling that account for the failures induced by the internal or external hazard events.

The scoping Seismic PSA used the seismic hazard evaluated at the Darlington site. The scoping Seismic PSA used generic bounding seismic capacities for SSCs that are set conservatively low. Using conservative SSC generic seismic capacities reveals those SSCs that contribute significantly to the seismic safety profile. With this information, the detailed design is performed demonstrating that sufficient hardening already exists (Chapter 9, Section 9B provides additional information regarding hardening) and provides focus for additional design enhancements to achieve the BWRX-300 safety goals through a risk-informed design process.

The internal events PSA analyzes and quantifies initiating internal events. Internal events generally include events occurring within the plant boundary, other than those involving internal hazards such as internal fires or floods. Historically, the internal events PSA includes the analysis of loss of offsite or preferred power, which has potential contributors both on-site (e.g., from the switchyard or switchgear) and offsite (e.g., grid or weather-related losses not involving other external hazards).

##### **15.6.1.2.1 Internal Hazards**

This section summarizes the screening for internal hazards for the BWRX-300 plant. Internal Hazards were screened for both consideration in the Probabilistic Safety Assessment (PSA) and for inclusion in the fault list. In addition, potential non-reactor sources of radioactivity are screened in this section.

From IAEA SSG-64, "Protection against Internal Hazards in the Design of Nuclear Power Plants," (Reference 15.6-11):

"Internal hazards are those hazards to the safety of the nuclear power plant that originate from within the site boundary and are associated with failures of facilities and activities that are under the control of the operating organization."

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

This section documents the screening of internal hazards for the BWRX-300 Standard Plant Design. Because the BWRX-300 PSA is an all-hazards PSA, internal hazards are evaluated for potential PSA development. NEDC-33946P, "BWRX-300 Darlington New Nuclear Project (DNNP) Probabilistic Safety Assessment Methodology" (Reference 15.6-25) provides guidance for screening external hazards; however, the screening approach may be adopted for the screening of internal hazards as well. The PSA Standard, ANSI/ASME/ANS RA-S-1.1-2022, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 15.6-12), Sections 1 and 6 provide several screening criteria.

The general approach screens internal hazards using the following steps:

1. Develop a list of candidate internal hazards.
2. Apply qualitative screening criteria to each of the internal hazards.
3. For those internal hazards remaining unscreened (i.e., pose a credible threat to nuclear safety) from Step 2, consider application of quantitative screening.
4. Any hazards left unscreened after Step 3 are retained as candidates for explicit treatment in the PSA.

In addition, on-site radioactivity sources are screened for potential treatment in the PSA.

### **Scope**

The internal hazard scope assessment applies screening criteria to all hazards originating within the site boundary. Several sources are consulted for developing of the internal hazards list. A list of on-site radioactivity sources is considered for treatment in the PSA. This subsection does not assess external hazards, sabotage/terrorism, nor is it used in the screening of external events.

Because the BWRX-300 uses a standard plant design and because internal hazards apply to those that originate within the site boundary, this screening section applies to both standard plant design and the DNNP.

### **Assumptions**

Administrative controls are implemented to preclude compressed gas cylinders from becoming missiles in areas containing risk-significant mitigating equipment.

1. Valves are designed to prevent removable parts from becoming missiles in the event of failure in accordance with guidance in SSG 64 (Reference 15.6-11).
2. Rotating equipment (excluding the main turbine) is designed such that potential failure-generated missiles are prevented from impacting risk-significant equipment through spatial or engineered means.
3. Administrative controls are placed to ensure stored combustibles are not collected in sufficient quantities to impact nuclear safety if ignited.
4. No risk-significant mitigating equipment resides in the RB Auxiliary Bay Plant Services Area truck bay or the service bay area beneath the truck bay.
5. The on-site radwaste system does not contain radioactivity in sufficient form or quantity to pose a public health hazard to the level of a small or large release as defined by the PSA.
6. Dry casks containing spent fuel, when stored or handled outside the cask pit, is capable of passive cooling and are not vulnerable to potential hazards.



## Hazard List

The preliminary list of BWRX-300 internal hazards is generated from industry guidelines, past studies, and a plant-specific review.

The internal hazards list of sources includes:

1. BWRX-300 Safety Strategy.
2. ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). This source is an aggregation of hazards based on review of industry studies such as USNRC NUREG/CR-2300, USNRC NUREG-1407, IAEA SSG-3, USNRC NUREG/CR-5042, EPRI 1022997, EPRI 3002005287, and ASAMPSA\_E List of External Hazards.
3. CSA N290.17-17 "Probabilistic safety assessment for nuclear power plants," (Reference 15.6-7).
4. IAEA SSG-3 (Reference 15.6-5), "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants".
5. IAEA SSG-64 (Reference 15.6-11), "Protecting against Internal Hazards in the Design of Nuclear Power Plants".

The initial hazards screening results are provided in Tables 15.6-1 through 15.6-2.

## Qualitative Screening

The following qualitative screening criteria are provided for internal and external hazards. Consideration of internal hazards is not fundamentally different than external hazards from a risk-impact perspective; therefore, these same criteria apply in screening internal hazards:

1. The event is of equal or lesser damage potential than events for which the plant is designed. This requires an evaluation of the plant design basis to estimate the resistance to a particular (internal) event.
2. The event has a significantly lower mean occurrence frequency than similar events included in the PSA and could not result in worse consequences.
3. The event cannot occur close enough to the plant to effect it.
4. The event is included in the definition of another event.
5. The event frequency is sufficiently low when compared to the probabilistic limits defined for the release category frequencies and is included in the PSA.

The following additional qualitative screening criteria are also considered:

1. The event does not result in a plant trip (manual or automatic) or require a plant shutdown.
2. The event develops slowly (DSA and PSA fault sequence figures show event progressions) and it is shown there is demonstrably conservative time margin available to eliminate the source or to provide adequate response.

The following events were assessed qualitatively:

- Release of chemicals from on-site storage
- Turbine-generated missiles
- Other internally-generated missiles from pressure vessels, valve failures, control rods

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- High speed rotating equipment (e.g., fan blades, turbines, pump impellers, fly wheels, coupling bolts)
- Internal fires are treated in the same manner as external events for BWRX-300 and are thus “screened-in” as part of this analysis alongside external hazards

Most explosions sources are within the Internal Fires PSA scope. Ignition of the following components with explosion potential is considered:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen sources
- Offgas/hydrogen recombiner
- Transformers
- Transient combustibles
- Boilers
- Turbine auxiliaries

Explosions are qualitatively screened based on Criteria 1.

The quantitative screening criteria from RA-S-1.4-2021 (Reference 15.6-2) are applied. A hazard screens quantitatively if:

1. Based on absolute risk contribution, an event sequence family subject to screening does not exceed the selected risk significance criteria and has mean occurrence frequencies less than  $1\text{E-}7/\text{plant-year}$ , as estimated using a bounding or demonstrably conservative analysis.
2. The total contribution of all screened out event sequence families does not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria.

The qualitative assessment of the above hazards is summarized in Table 15.6-1.

### **Quantitative Screening**

The quantitative screening criteria for external hazards are also used for internal hazards.

To determine the CDF from turbine-generated missiles, the IE frequency, and Conditional Core Damage Probability (CCDP) are determined.

The CCDP was calculated by assuming a LOPP and failure of diesel generators and was estimated to be approximately  $3.4\text{E-}7$ . Multiplying the CCDP by the IE frequency,  $1\text{E-}4/\text{yr}$ , yields a conservative CDF of  $3.4\text{E-}11/\text{yr}$ . The  $1\text{E-}4/\text{yr}$  frequency is conservative because it does not credit the fractional probability of the turbine missile striking and then damaging the target equipment.

Turbine-generated missile hazards meets the screening criteria, so this hazard screens quantitatively.

Internal flooding hazards are treated explicitly in the PSA. No quantitative screening is performed for internal flooding.

The results of the Internal Hazard quantitative screening analysis are presented in Table 15.6-2.

## **Hazard Combinations**

### *Combinations of External Hazards and Correlated Hazards*

The combinations of external hazards or correlated hazards (referred to below as combinations of hazards) are analyzed for those hazards that have the potential to combine, e.g., heavy winds causing the loss of the external grid, snowstorm drifts blocking the diesel generator combustion air intakes and frazil ice blocking water cooling.

### *Categories of Hazard Combinations*

**Consequential Hazards:** Combinations of consequential (or subsequent) events are defined consistent with IAEA, SSG-64, Appendix I (Reference 15.6-11):

An initial event, e.g., an external or internal hazard, results in at least one further hazard, e.g., another external or internal hazard. Event chains of three or more hazards subsequent to each other are possible.

For the assessment of external hazards combinations, the initial event impacting a nuclear installation is either a natural hazard such as an earthquake or an external flooding, or a human-induced one, such as an accidental aircraft crash or an explosion pressure wave (blast) originating from outside the plant boundary.

The following hazard combinations involving external hazards are possible to occur:

1. Natural hazard and consequential external hazard covering:
  - a. Natural hazard and consequential natural hazard (e.g., earthquake and consequential tsunami)
  - b. Natural hazard and consequential human-induced hazard (e.g., extreme wind and consequential aircraft or helicopter crash)
  - c. Natural hazard and consequential internal hazard (seismic and fire)
  - d. Human-induced hazard and consequential internal hazard (aircraft crash and internal explosion)
2. Correlated Hazards: Combinations of correlated hazards are defined consistent with IAEA, SSG-64, Appendix I (Reference 15.6-11):
  - a. Two or more external and/or internal hazards occur as a result of a common cause initiator. The common cause can be any anticipated event including an external hazard or might be due to an unanticipated dependency. The two or more hazards correlated by this common cause could occur simultaneously.
3. The following hazard combinations involving external hazards may occur:
  - a. A natural hazard or hazard source as common cause for other external and/or internal hazards and/or internal events. Typical examples include an earthquake as the common cause for landslide, external fire, internal fire, internal explosion, and station blackout, as potentially correlated events, or tsunami as the common cause for external flooding, internal flooding, and internal fire as potentially correlated events.
  - b. A human-induced hazard (or hazard source) as common cause for other human-induced and/or internal hazards. Examples are an accidental aircraft crash of a large commercial passenger aircraft as the common cause for an external

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

explosion (outside the plant boundary), a plant internal fire an internal explosion, and a heavy load drop correlated by the aircraft crash, or an electromagnetic interference outside the plant boundary as the common cause for a plant internal electromagnetic interference and internal fire as the two correlated hazards.

4. Independent Hazards: Combinations of unrelated (or independent) events are defined as follows consistent to IAEA, SSG-64, Appendix I (Reference 15.6-11):
  - a. An initial event (e.g., an external or internal hazard) occurs independently from, but simultaneously with another hazard without any common cause. The term 'simultaneously' in this case does not mean that the hazards occur exactly at the same time but rather that the second hazard occurs before the consequences of the previous hazards have been completely mitigated.
5. The following event combinations involving external hazards may occur:
  - a. Natural hazards and independently occurring natural hazards. Typical examples include longer duration riverine flooding and independently occurring extreme weather conditions, such as extreme wind.
  - b. Natural hazard and independently occurring human-induced hazard. Examples are a longer duration hydrological hazard and an independently occurring industry accident (such as an explosion, release of hazardous substances, etc.).

*Identification and Screening of Hazard Combinations*

Event combinations used to develop a list of potential combined hazards is initially based on the identified individual hazards. The number of individual hazards identified to occur at a given plant location is already high. Building combinations of all possible individual hazards result in a number of combined hazards, which is not manageable without applying a systematic approach.

Identification of consequential hazards begins with the unscreened single hazards and determining those hazards that may occur as a result of the initial hazard. This can be performed using a matrix approach, listing the hazards graphically and showing the combinations of two hazards (so-called first order combinations) where the possible combinations are identified. The combined hazards can be further reviewed based on this initial identification to determine additional hazard combinations possible (second-order combinations) or additional identified combinations.

For combinations of correlated hazards, it is not directly possible to build a matrix for these combinations. The analyst starts the combined hazard screening by identifying potential common causes for those individual (single) hazards not screened out for the plant/site and builds a tree-type structure (or similar) for all hazards correlated by such common causes representing the roots of the branches. With such a tree-type structure, higher order combinations of correlated hazards can be built and undergo a screening process.

For the third category of combinations of independent hazards, the same two-dimensional matrix structure is used. The possibility of an event combination is demonstrated by a color change or other designation in the corresponding matrix field. The matrix involves the identification of hazards that may impact the plant/site for a time period longer than 3 days. The likelihood is included in the matrix based on the duration of the first hazard and the likelihood of the second hazard occurring during that time period. Combined events are identified by analyzing the correlations and the effects on the plant. The analysis of possible correlations (dependency) between events is made by assessing the physical bases of the phenomena, observed data, actual operating events, and general knowledge of local conditions.

Expert judgment and rough quantitative analysis are used for estimating correlations. The observed data for intensities relevant to external events PSA is sufficient for order of magnitude correlation estimates.

For example, high wind often results in high water level and organic material in water (e.g., seaweeds, kelp, algae, Asiatic clams, mussels, etc.). The combination may simultaneously endanger offsite power and the diesel generator cooling water intake; therefore, it is analyzed as a combined event.

Extreme wind velocities are measured in winter and could be associated with snowfall resulting in possibility of simultaneous LOPP and loss of the diesel generators due to snowdrift blockage of combustion air intake.

Simultaneous high air and seawater temperature could also endanger plant equipment, e.g., instrument room cooling at units with diverse cooling system heat sinks. Identified hazard combinations are screened similar to single hazards using the process discussed previously.

#### *Evaluation of Risks Associated with Unscreened External Hazards*

For unscreened external hazards and identified combinations of external hazards, a similar model development and quantification processes described previously are adapted to evaluate external hazard risks. The process utilized for hazards combinations is discussed below.

Several external hazards from transportation and nearby facilities are evaluated. These external hazards are evaluated with bounding assumptions and can be modified with plant-specific inputs:

- Airports and airways hazards
- Industrial accidents
- Pipeline accidents
- Hydrogen storage failures
- Transportation accidents

#### *Process for Selection, Screening and Analysis of Combinations of Hazards*

Figure 15.6-2 provides an overall process for selecting, screening, and analyzing combinations of hazards. The quantification of the hazards is performed similar to the quantification of the individual hazards with consideration of the combined hazard on the SSC fragilities, the likelihood of the combined hazards, and the inclusion of these factors in the PSA model.

The process for the specific modeling depends on the hazards, and the supporting single hazard PSA modeling. During the PSA quantification, the quantification steps may be slightly different if the combined hazards PSA results are calculated together with or separately from the single hazards PSA model quantification. For example, if a seismically-induced flooding event PSA model is developed and quantified, it can be quantified as a part of the Seismic PSA or separately where only the accident sequences involving seismically-induced flooding are quantified. There are no requirements in either IAEA-TECDOC-1804, "Attributes of Full Scope Level 1 Probabilistic Safety Assessment for Applications in Nuclear Power Plants" (Reference 15.6-13) or the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) to quantify single and combined hazards separately or together. There are advantages of long-term model maintenance in developing and analyzing the combined hazards using a single model. The quantification of each single and combined hazard avoids double counting the results. For example, if the CDF of a single hazard PSA model such as a seismic PSA provides a value of  $1\text{E-}05$  / reactor year, and the combined hazards PSA model for the combination of consequential hazards seismically-induced internal flooding provides a CDF of  $5\text{E-}06$ /reactor year, there is likely some double counting if the final

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

results are equal to the sum of the two CDF values. In this case, a realistic assessment of the single hazard PSA is based on accident sequences where a flooding event does not occur.

The quantification of the combined hazard PSA is performed in a similar manner as the individual hazard PSA, accounting for the conditional probability/frequency of the combined hazard magnitudes. As previously discussed, this can be applied in a variety of approaches, but often specified magnitude ranges or bins are used. The analysis of hazard combinations magnitude ranges or bins include variation of the various hazards and the conditional probability of each hazard in combination with the others.

### **Internal Hazard Combinations**

Heavy load drop-induced floods and structure damage are addressed as part of the heavy loads analysis in the low power/shutdown PSA evaluation.

Turbine-generator missile-induced fires/explosions are addressed in the fire PSA. Pipe whip and jet effects are addressed in the internal flooding PSA.

### **External Hazard Combinations**

The external hazards combinations were identified and screened. The high wind combinations are covered by the high wind PSA.

The BWRX-300 PSA scope includes considerations of radioactive sources outside the core. Since one goal of the PSA is to ultimately evaluate the level of safety to the surrounding populace, only those radioactive sources containing sufficient radioactivity that, if released, would pose a significant health hazard outside the site boundary.

The only significant radiation source in the primary containment is from the reactor core. The reactor core is treated in the PSA. The FMCRD system also contains a radiation source but is not expected to pose an offsite threat.

In the RB, the fuel pool is assumed to contain during some portion of the plant lifetime sufficient radioactivity to pose a public health hazard outside the site boundary. This radiation source is treated in the PSA.

Radioactivity deposited on systems in the plant is not liberated in sufficient quantities to pose a public health hazard. This includes systems such as the fuel pool cooling, reactor water cleanup, main steam, feedwater, and Heating, Ventilation, and Air Conditioning (HVAC) systems.

The radioactive waste system does not contain radioactivity in sufficient form or quantity to pose a public health hazard in the form of a small or large release defined by the PSA.

The potential radionuclide source term from severe damage to spent fuel casks was examined in USNRC NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants." (Reference 15.6-14). USNRC NUREG/CR-6451 (Reference 15.6-14) considered tornado missile impact the limiting hazard and the tornado wind speed required to generate a missile with enough force to compromise a cask has never been observed. The judgment that tornado impacts are the limiting physical hazard to spent fuel dry casks is adopted for the BWRX-300 design. Regardless of the ruggedness of the casks, a conditional radionuclide source term was developed for a cask event where all rods of a fuel assembly are compromised. The release fractions for iodine and cesium are less than  $1.5\text{E-}5$  and  $2.25\text{E-}5$  respectively, from the damaged spent fuel assembly in the cask. The Level 2 Probabilistic Safety Assessment estimates that the release fraction for I-131 constitutes a small release of  $1.14\text{E-}3$  and release fraction for Cs-137 constitutes a large release of  $5.20\text{E-}4$  in terms of fractional inventory of the entire core.

Because hazards to spent fuel in dry casks do not result in fuel damage or significant radionuclide releases, dry casks are screened from treatment in the PSA.

#### **15.6.1.2.2 External Hazards**

The BWRX-300 PSA is an all-hazards PSA where external hazards are evaluated for potential PSA development.

The methodology examines the hazards from Appendix 2 “Questionnaire on External Events PSA (Other Than Seismic) of NEA/CSNI/R(2009)4,” “Probabilistic Safety Analysis (PSA) of Other External Events Other Than Earthquake,” (Reference 15.6-15) and applies screening to each hazard according to the rules of ANSI/ASME/ANS RA-S-1.1-2022, Appendix 2 (Reference 15.6-12):

1. A hazard qualitatively screens if demonstratively conservative assessments find the hazard does not impact the plant or is subsumed into a more frequent or more impactful event. Any one of the following qualitative screening criteria may be used:
  - a. The event is of equal or lesser damage potential than events for which the plant is designed. This requires an evaluation of the plant design basis in order to estimate the resistance to a particular (external) event.
  - b. The event has a significantly lower mean frequency of occurrence than similar events included in the PSA and could not result in worse consequences in those events.
  - c. The event cannot occur close enough to the plant to effect it.
  - d. The event is included in the definition of another event.
2. The frequency of the event is sufficiently low in comparison with the probabilistic limits defined for the release category frequencies so that it does not need to be included in the PSA. Hazards that do not screen qualitatively are subjected to quantitative screening. A hazard screens quantitatively if:
  - a. Based on absolute risk contribution, an event sequence family subject to screening that does not exceed the selected risk significance criteria and has mean occurrence frequencies less than  $1\text{E-}7/\text{plant-year}$ , as estimated using a bounding or demonstrably conservative analysis, and.
  - b. The total contribution of all screened out event sequence families may not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria
3. Hazards that screen neither qualitatively nor quantitatively are addressed by detailed PSA. A decision can be made to bypass screening of a hazard and go directly to detailed PSA development.

The NK38-REP-03611-10043 R003, “Hazards Screening Analysis – Darlington” (Reference 15.6-26) site environmental conditions were used for external hazards screening:

1. For the screening assessment of high water level, a 100-year flood is conservatively assumed to flood the site and allow water to enter the RB.
2. The BWRX-300 Diesel Generators (DGs) are air-cooled, and thus operation may degrade or cease in extreme ambient temperature conditions.
3. Given the advent of weather extremes due to global climate change, an extreme high temperature condition of  $37.8^{\circ}\text{C}$  or more is assumed to occur at the site once per 20 years.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

4. Safety SSCs exposed to ambient environmental conditions and required to function in local extreme meteorological events are assumed to be designed for -40°C.
5. For extreme high or low ambient temperatures, a LOPP is assumed to occur due to high energy demand.
6. The DGs, which are air-cooled, are assumed to be unavailable given extreme temperatures at or below -40°C that may lead to station blackout.
7. A northern climate, Toronto Canada, is selected as a representative location for snowfall at the Darlington site. Historical heaviest single day snowfall data for Toronto for the past 169 years indicate that the maximum snowfall episode was 48.3 centimeters on December 11, 1944. Heavy snowfall did not extend beyond or prior to this single day. It is assumed that 48 centimeters is below the design snow load for non-Seismic Category-I (non-SC-I) structures and below the air intakes for the DGs.
8. It is assumed that if heavy snowfall is forecast for multiple days, operator action will initiate shutdown procedures if the integrity of plant structures due to heavy loading is threatened.
9. BWRX-300 structures are designed to withstand credible accumulations of hail stones.
10. It is assumed that all outdoor structures of the BWRX-300 are designed to withstand white frost.
11. It is assumed that the plant is not susceptible to sand accumulation that may affect SSCs.
12. The BWRX-300 plant design protects electrical systems functions from voltage transients, including those caused by lightning.
13. For explosions, it is assumed that the Darlington BWRX-300 site is situated beyond the safe screening distance for explosion 700m and vapor cloud explosion 460m to the Saint Mary's Cement Plant.
14. If an explosion occurs in the vicinity of the site, a loss of all structures other than the reactor building is assumed including the MCR as well as LOPP. The DGs are assumed to remain available as they are in a bunkered area and air-cooled (assumed). Electrical cables and control cables are assumed to not be impacted (e.g., they run underground).
15. For marine transportation accidents, a restricted zone is assumed to exist surrounding the plant intake channel and diffuser that does not allow commercial ships to approach the site shoreline. Therefore, marine vessel explosions will not harm the site.
16. It is assumed that the Darlington site is a safe distance from blasting.
17. The Darlington site is assumed situated beyond the screening distances for explosions after pipeline accidents.
18. Chemical release can occur due to the rail line accidents near the site. The control room HVAC has design provisions to address a chemical release.
19. It is assumed that the site is situated beyond the screening distances for chemical releases after pipeline accidents, and chemical releases to water. There are three pipelines (2 natural gas and one petroleum) that are located beyond the screening distance for the Darlington CANDU reactors.
20. BWRX-300 I&C systems are designed to prevent magnetic disturbance from significantly impacting them.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

21. The RB is designed to withstand large aircraft impact. The RB utilizes steel bricks construction, and the design parameters for the Steel Bricks™, such as steel thickness, steel ductility, and concrete thickness, is selected to withstand RB perforation from aircraft and aircraft engine impact.
22. The potential exists for a direct hit from an aircraft engine to one of the RB exterior doors and subsequent penetration. The CCDP given for an airplane crash is 1.0.
23. It is assumed that the BWRX-300 at Darlington site includes an evaluation that demonstrates no historical evidence of subsidence.
24. It is assumed that BWRX-300 building foundations are designed to address potential soil frost and that pipes are buried below the frost line.
25. The Darlington site is not located near volcanoes. There is no concern for ash to fall on the plant.
26. Darlington topography is relatively flat surrounding the site. No avalanche potential exists.
27. The Darlington site includes consideration of waterline topography with relatively flat topography.
28. The land coverage surrounding Darlington is expected to have minimal vegetation, limited to bushes, and mostly paved in the areas proximal to site buildings. It is assumed there is limited numbers of trees, and that the area is mostly paved and cleared to preclude the possibility of external fires or tornado missiles, causing damage to equipment, or impacting control room operations. As a result of the sparse vegetation, external fire propagation to the site is not credible.
29. It is assumed that plant procedures are in place to preclude damage to underground equipment, such as piping or tanks, from excavation work.
30. It is assumed that plant procedures are in place to preclude damage to underground equipment, such as piping, from heavy transportation within the site.
31. Missile impacts from military activity are assumed to be bounded by the airplane impact hazard.
32. It is assumed that the Darlington BWRX-300 plant will not be contiguous with another existing plant.
33. Lake Ontario that is part of the Darlington site boundary is assumed not to have strong lake water currents.
34. Low water level slowly develops. It is assumed that the plant would be shut down and reach a safe stable state prior to the loss of circulating water or plant cooling water.
35. It is assumed that the Darlington BWRX-300 site waterline topography allows water to drain away from the plant.
36. Surface ice will not affect plant operation. The coolant intake and discharge lines at the intake structure are assumed to be located on the water body floor.
37. It is assumed that the Darlington BWRX-300 site is located a safe distance from shipping lanes.
38. For quantitative screening in which offsite power is lost, the CCDP calculations assume that the offsite power loss is not recovered that is used in the internal events PSA modeling.

OPG has used drones for the Darlington CANDU Vacuum Building inspections. The impact of drones hitting the BWRX-300 Structures Systems and Components (SSCs) is bounded by small aircraft crash.

The USNRC has reviewed impact of drones on U.S. Nuclear Power Plants (Backgrounder – Drones and Nuclear Power Plant Security (Reference 15.6-16)). Their assessment states:

“The technical analysis concluded that U.S. nuclear power plants do not have any risk-significant vulnerabilities that could be exploited by adversaries using commercially available drones to result in radiological sabotage, theft, or diversion of special nuclear material (essentially the reactor fuel)”. Thus, drones are screened out.

## **Results**

The external hazards qualitatively screened out from the PSA are shown in Table 15.6-3. The external hazards quantitative assessment results are shown in Table 15.6-4.

### **15.6.1.3 Level 1 PSA**

#### **15.6.1.3.1 Level 1 PSA At-Power**

One of the first and basic steps in a PSA is the identification and quantification of the IEs used in the sequence analysis. IEs have historically been broadly classified as either “internal” or “external” hazards. An IE may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, floods, or fires), or external to the plant (e.g., earthquakes or high winds), or combinations thereof. Internal events PSA assesses events that are caused by systems or components located within the site structures. An exception of this is LOPP that is analyzed in the internal events PSA. Internal and external hazard induced IEs (e.g., seismic events, internal floods) and IEs during shutdown excluding sabotage are discussed in other sections in this report:

- Probabilistic Fire Analysis (internal fire events)
- Probabilistic Flood Analysis (internal flooding events)
- High Wind Risk (including tornado events)
- Seismic PSA (seismic events)
- Shutdown Risk (shutdown IE for internal events)
- Spent fuel damage

#### **15.6.1.3.2 Initiating Events Identification**

To develop a comprehensive accounting of postulated IEs, a systematic approach is used to identify events that challenge normal plant operation and require successful mitigation to prevent radionuclide release. IAEA-TECDOC-1804 (Reference 15.6-13) provides the following IE definition: “an event which could directly lead to core damage or challenges normal plant operation and requires successful mitigation to prevent core damage”.

A comprehensive identification of IEs includes a review of existing BWR PSAs, including the ESBWR, and generic sources. Generic lists of IEs are included in multiple reference documents such as IAEA-TECDOC-719, Purpose of Probabilistic Safety Assessment,” (Reference 15.6-17), USNRC NUREG/CR-5750 and USNRC NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants.” (Reference 15.6-19) update 2015.

IEs identified typically include transients of various types, LOCAs, and support system initiators. IEs under low power conditions are also identified and are listed in Table 15.6-5.

All system failures that disturb plant operation (e.g., front-line and support systems) are reviewed, except for those already identified as the source IE based on other analyses.

In addition, planned and unplanned manual shutdowns which seldom place demands on any standby safety equipment are treated as IEs because of their high frequency and because they represent changes in operating states that result in demands on available equipment to reach a safe shutdown condition.

Support system initiators are defined as unplanned normal shutdown resulting from loss of a support system with dependent failures. Related to support system initiators are CCF that are equipment failures that cause an IE and simultaneously degrade equipment credited for mitigating the IE (e.g., failure of balance of plant equipment that could cause a plant trip and preclude use of the normal power conversion system). CCFs are systematically identified and treated in the PSA.

Human error IE is considered for these analyses.

An example list of the general damage definitions of internal IE are shown in Table 15.6-6, which is taken from IAEA-TECDOC-719, Table 3.9 (Reference 15.6-17). This list is supplemented with the other generic sources previously discussed in developing a comprehensive IE list.

#### **15.6.1.3.3 Initiating Events Screening**

The requirements for screening potential IE candidates from the PSA are discussed in IAEA-TECDOC-1804 (Reference 15.6-13) and the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

Based on the references in Subsection 15.6.1.3.2, the IE candidates identified are excluded from further consideration if they meet one of the following criteria:

1. The event does not lead to the IE as defined in the PSA.
2. The event does not correspond to the scope of the PSA.
3. The frequency of the event is less than the truncation value related to the accident sequence frequency, and the event does not involve an interfacing systems LOCA, containment bypass, or RPV rupture. For these events, the truncation value is at least one order of magnitude lower than the truncation value accepted in the PSA.
4. The resulting reactor shutdown during at-power is not an immediate occurrence. The event does not require the plant to transfer to shutdown conditions until a defined amount of time has elapsed, the condition is detectable before plant systems are required to respond, and there is a high degree of certainty (based on supporting calculations) that the condition can be detected and corrected before normal plant operation is affected (either administratively or automatically).

The IE screening is limited to those events that do not lead to an IE defined in the PSA, and those events where a shutdown occurs prior to the conditions being corrected. Screening is carefully and conservatively applied, especially early in design where the impact for an event may not be fully understood without design details that are fully developed.

#### **15.6.1.3.4 Accident Sequence Analysis**

The event trees developed for BWRX-300 are based on the IEs developed in the IE analysis. The event tree models include the set of safety functions needed to mitigate each IE. The accident

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

sequence analysis ensures that the response of the plant systems and operators to an IE is reflected in the risk assessment so that:

1. Operator actions, mitigation systems, and phenomena that influence or determine the course of sequences are appropriately included in the accident sequence model and sequence definition.
2. Plant-specific dependencies due to IEs, human interfaces, functional dependencies, environmental, and spatial impact, and CCFs are reflected in the accident sequence (event tree) structure.
3. The individual function successes, mission times, and time windows for operator actions for each critical safety function modeled in the accident sequences reflects the success criteria evaluated in accordance with the acceptance criteria attributes following.
4. End states are clearly defined as either core/fuel damage or successful prevention with the capability to support the interface between Level 1 and Level 2 PSA.
5. The accident sequences are defined for the selected set of IEs, POSs, and times that a POS can occur.

The resulting accident sequence analysis addresses:

- General assumptions relating to all event tree development
- Sequence end states
- Definition of the type of models produced (e.g., small event trees and large fault trees) and level at which the event tree headings are defined (safety function, system, train)
- The BWRX-300 event trees are generally small event trees developed using the CAFTA software discussed previously. Event tree headings are largely based on top-level system functions with functional fault trees added to connect the models to system and train failure events
- Requirements for developing event sequence diagrams
- Interface between the IE, success criteria, human reliability, system modeling and data analysis tasks

The event sequence analysis meets the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

## **Methodology**

### *Acceptance Criteria*

A formal definition of core damage is defined and justified before the function overall success criteria is established.

Core damage is defined in terms of physical processes, phenomena, and failure mechanisms. The accident analyses and past experiences for similar plants are used in deriving the core damage definition. The physical plant parameters (e.g., highest node temperature) and associated acceptance criteria limit values (e.g., percentage of cladding thickness oxidized) used in determining the core damage definition. The parameters selected in determining core or fuel damage is practical, consistent, and realistic. Best practices and OPEX, conservatism in computer codes, sophistication of models, and sufficient margin for uncertainties are parameters considered in defining core or fuel damage.

The success criteria supporting each event tree and key safety functions are the minimum requirements necessary to achieve safe, stable conditions (i.e., to protect the fuel and prevent

radionuclide releases to the environment). Safe conditions are determined by meeting the following key safety function success criteria, and stable conditions are determined by maintaining each key FSFs for long-term operation:

1. Reactivity Control: Achieve subcriticality and maintain the reactor in a subcritical state (combined with adequate core cooling and/or containment heat removal functions) so that core damage and containment failure are avoided.
2. Core Cooling: Adequate core cooling is provided to prevent core damage and maintain the reactor in a safe-and-stable condition. An example of the overall success criteria for core cooling is ensuring the PCT does not exceed 1200°C and cladding oxidation does not exceed 15%.
3. Containment System Integrity: Ensure that the containment pressure does not exceed the ultimate containment failure pressure, and containment isolation occurs when required.

Success in achieving the FSFs is defined in Chapter 15, Section 15.2.

Other safety functions are developed as needed. For example, the RCPB success criteria is developed and modeled based upon the analyzed maximum pressure increase in the reactor coolant.

### **Accident Sequence Development**

The role of an event tree is graphical expression of IE groups, success criteria, and sequence end points. The basic concept is preparing an event tree for each IE group defined in IE analysis task.

IE group is studied and organized based on following policy based on IAEA-TECDOC-1804 (Reference 15.6-13): "For each IE group for internal events, internal hazards, and external hazards for each POS the accident progression for all sequences is identified and justified. For each IE group the accident sequence models are developed. As models explicitly address realistic plant behaviour in response to IE in terms of normal plant systems operation, operator actions, and mitigation systems that support the key safety functions necessary to achieve a stable safe state."

### **End States of Event Sequences**

The event trees identify the potential sequences that may lead to radionuclide release. Many of the sequences have common characteristics that challenge the containment fission product barrier. These sequences are grouped into damage classes that are analyzed in the release portion of the PSA. The end states of the event sequences developed are defined to facilitate the containment performance analysis and provide the link between core damage and a release category.

Event sequences are then qualitatively analyzed to determine which sequences lead to core damage end states.

The core damage sequences are grouped together based upon the overall challenge to the containment barrier and defined as:

1. OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
2. CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.

3. CD II: The containment is breached, either due to overpressurization or venting, while the core is successfully cooled. Core damage results from failure of long-term heat removal core cooling.
4. CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.
5. CD IV: Core damage results from an accident sequence with initial failure of effective reactivity control (CCF failure of hydraulic scram or FMCRD run-in, control rod binding, etc.). This potentially affects containment more severely than CD I or CD III because more energy is deposited into the containment prior to RPV failure. All CD IV end states are treated as CD I or CD III depending on the RPV pressure without affecting the results of the containment analysis. This end state has been retained in this scoping analysis to readily allow sensitivity analyses for reactivity control.
6. CD V: The containment is bypassed at the time of core damage.
7. CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

Sequences where core damage is expected to occur very late are appended with an 'L'. An example of this occurrence is when CRD flow is being relied upon to avoid core damage, but eventually fails due to inventory depletion.

#### **15.6.1.3.5 Success Criteria Analysis**

The main objective of success criteria formulation is determining for given IEs what represents a successful or unsuccessful plant response and translating this information into detailed plant system and operator action success criteria accomplished for at-power or low power/shutdown operation. The BWRX-300 success criteria analysis is integrated with the accident sequence analysis.

Success criteria are defined for several different levels: The highest level of success criteria development determines what mitigates an IE (i.e., aversion of core damage). Using the qualitative description of core damage (e.g., gross fuel/cladding degradation due to failure to adequately remove fission and/or decay heat), a parametric description (e.g., peak core nodal temperature) generated.

Success criteria are developed supporting the functions required to mitigate core damage. An assessment of front-line systems identifies mitigating functions and determines the requirements to fulfill those functions. Best estimate analyses use computer codes or other acceptable calculations in performing this task. Examples include determining how many trains of a system performs a function for IE sub-groups.

Supporting success criteria is determined once the functional success criteria is identified. This includes mission times calculations and human actions, inventory availability determination, and other system or train-level success criteria development.

### **Methodology**

#### ***Definition of Core Damage***

The definition of core or fuel damage is a key aspect in determining the functional success. Core damage is defined by the two key physical parameters: PCT and fuel cladding oxidation.

The above definition and criteria are consistent with the definition in PSA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) as "uncovery and heat-up of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects."

### *Functional Success Criteria*

The functional success criteria defines the minimum performance that differentiates the end state when that function succeeds from the end state reached when the function fails. With the parameters that determine core damage and decay heat levels defined, specific success criteria are added defining the necessary capability of SSCs and operator actions in accident sequences.

A mission time is determined to achieve a stable end state after an IE for the accident sequences. As a first approach, a general mission time of 24 hours is assumed for Best Estimate (BE). A longer mission time is used for some accident sequences to achieve stable plant conditions.

Success criteria analysis considers the available inventories needed to support each system or function. The available inventories of fuel, water, or air in tanks etc., are compared with those required to support each success criterion and the model reflects the comparison results.

Analysis models and computer codes are used to model the conditions and phenomena that determine the success criteria. The plant model and parameters used for thermal-hydraulic analyses provides sufficient resolution and reflects the actual design and operational features of the plant.

For each accident sequence condition, the applicability ranges and conditions for each analysis code are reviewed. An assessment is performed ensuring that all accident sequences are within the applicability of the codes used in determining the success criteria. The use of all codes is justified based on this comparison.

Each success criterion in the analysis is based on either a plant design parameter or a thermal-hydraulic calculation. When a success criterion is used to characterize a range of conditions, the limiting parameters are chosen to represent all cases. In addition, representative thermal-hydraulic analyses and offsite dose consequence analyses are run to validate the appropriateness of the success criteria.

#### **15.6.1.3.6 System Analysis**

The system analysis is performed for each plant system represented in the IE analysis and accident sequence analysis for each POS in such a way that:

1. Each safety function in accident sequence models and system models are developed that account for the success criteria.
2. System-level success criteria, mission times, time windows for operator actions, different initial system alignments and assumptions provide the basis for the system logic models reflected in the model. A reasonably complete set of system failure and unavailability modes for each system is represented.
3. Human errors and operator actions that influence the system unavailability or the system contribution to accident sequences are identified for development as part of the HRA element.
4. Intra-system dependencies and intersystem dependencies including functional, human, phenomenological, and CCFs that influence system unavailability or the system contribution to accident sequence frequencies are identified.

Other objectives include:

- Considering credible failure modes
- Modeling each failure mode impact on system performance
- Including support system failure modes in the front-line system fault trees

- Creating linked fault tree models that allow resolution
- Creating fault tree models where cutsets are clear and concise

## **Methodology**

The BWRX-300 system analysis includes systems modeled in the BWRX-300 PSA, which correspond to the functional headings described in the event sequences plus any support systems needed to accomplish those functions. The system modeling methodology follows the guidance in USNRC NUREG/CR-2300, USNRC NUREG/CR-4550 and USNRC NUREG/CR-4551 series. USNRC NUREG-0492 provides additional guidance on fault tree modeling, current information for the system studies, and latest results.

The analysis for each system begins by defining the system functions, system operation modes and alignments, establishing system-level success criteria (in conjunction with the event tree development that addresses operability and survivability), and defines the system boundaries. All system components and system connections that are modeled are described in simplified drawings.

For systems that operate in several different modes or POSs, each mode and the functions performed, are explained. Only those functions that contribute to plant risk defined by the accident sequences are included. The interfaces and system connections of the system with other systems are described. This also includes the support systems and relevant system components required for the operation of the system. Any other relevant aspects of the system operation including system trips, interlocks and procedural restrictions during accident conditions are also addressed.

Necessary operator actions are identified for systems where manual initiation is necessary or manual backup is credited in the PSA. The post-initiator HFEs (Type C) are identified based on operational procedure. Significant pre-initiator HFEs (Type A) are identified by investigating test and maintenance procedures related to the SSCs modeled in fault trees and are identified as part of the system analysis.

The success criteria of front-line systems for each IE are identified in the success criteria analysis task. System success criteria of support system are obtained by following success criteria for front-line systems. Based on these success criteria, necessary system functions are identified. Operability relative to trip setpoint limits and adverse environmental effects impacting all system components survivability are addressed. Any operator interventions that mitigate adverse environmental effects is proceduralized.

Systems models are developed that include all component failure modes and unavailability factors that lead to failing to achieve the system function defined by the system success criteria. Component failures that would be beneficial to system operation are not included in the model.

Database for component failure probability has two general types of failures. One is a demand failure, and the other is a rate failure based on the amount of time a component needs to operate. The reliability models considered to support BEs are not discussed in detail here, but involve exposure-rate for demand failures, exposure-rate or periodic test models for rate failures, and models that account for monitoring and repair. The data that supports the parameters listed are discussed in the data analysis task.

The documentation of the system analysis describes the processes used in modeling and quantification, generic modeling assumptions and is constructed considering future updates and uses in risk-informed applications.



#### **15.6.1.3.7 Fault Trees**

One fault tree for each system function is included in the PSA model. If multiple trains perform the function, then each train is modeled. All the logic for a particular system is contained in a single fault tree file.

The fault trees are constructed using the gate and basic event naming conventions. The fault tree database is stored in the CAFTA database file format.

#### **15.6.1.3.8 Data Analysis**

The data analysis objective is providing estimates for the reliability parameters for the reliability models specified under the systems analysis. The reliability models are used in determining the BEs probabilities of specific equipment failures and unavailability factors.

Data analysis is used in deriving the parameter values that estimate system failure probabilities and accident sequence frequencies. These BEs used in the PSA model are outputs from the IE analysis, accident sequence modeling, and system fault tree modeling discussed previously. The PSA BEs include the IE frequencies, component unavailability, component/train unavailability factors (resulting from testing or maintenance), CCF events, and other types of events. At this stage in the design process, there is no plant-specific reliability data available. Generic data from the nuclear industry and associated uncertainty are used.

Where plant-specific reliability data are not available, mainly generic reliability data are used as discussed below under component reliability database.

In the absence of actual operating experience for the passive plants, Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) recommends using failure data for components that are most similar to those used in passive plants. Additional adjustments to the generic data are introduced after analyzing the test and maintenance intervals and the environmental factors.

The data and basis for test and maintenance unavailabilities are based on bounding generic values and OLC allowed outage times. Because of limited operating experience and the lack of plant-specific data, the development of failure rates for equipment reflects appropriate characterizations of the associated uncertainties.

The level of redundancy in passive plant design and few critical support systems leads to increased focus on CCFs. CCFs are modeled for the components of the same size and in the same operating environment. The CCF data from the USNRC CCF (Reference 15.6-18) is used in this analysis. Generic CCF factors are used when component-specific data are not available.

### **Methodology**

#### ***Component Reliability Database***

The parameters are estimated using data and other information that is compatible with the definitions of the BEs. The task procedure either contains the definitions of the events or does so by task procedures for systems analysis and IEs. The component boundaries and the interfaces among connected components are identified. Because a component is required for different operating conditions, BEs representing different failure modes is included in the system model.

Components are grouped into population groups for parameter estimation. The rationale for grouping components into a homogeneous population for parameter estimation considers the design, environmental, functional, and operational conditions of the components in the as-built

and as-operated plant. For parameter estimation, components are grouped according to type and to the detailed usage characteristics.

In this analysis, generic reliability data-based USNRC NUREG/CR-6928 (Reference 15.6-19) are used. When data from this USNRC NUREG/CR-6928 (Reference 15.6-19) is not available for a modeled failure mode, data is used from the following resources in this order:

- EPRI TR-016780-V3R8, "T-Book: Reliability Data of Components in Nordic Nuclear Power Plants, 6th Edition"
- USNRC NUREG/CR-4550, "Analysis of Core Damage Frequency From Internal Events: Methodology Guidelines"
- USNRC NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide"
- USNRC NUREG/CR-1740, "Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants"
- EGG-SSRE-8875, "Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs"
- Engineering judgment

### **Dependent Events and Common Cause Failure Analysis**

Dependent events challenge redundancy or diversity and ultimately increase the unavailability of a system. Four dependency types are analyzed in the PSA analysis:

1. Plant-level functional dependencies: Plant-level functional dependencies lead to functional failures as a direct result of a shared IE.
2. Intersystem dependencies: The dependencies of BWRX-300 systems are carefully analyzed in the PSA analysis. Physical and functional intersystem dependencies are incorporated into the system fault trees by using transfer gates. Physical dependencies include spatial interactions, influences for common environment, or impacts from adjacent SSCs. These physical dependencies are important when modeling the impacts from internal and external hazards( fire and flood).
3. Human action dependencies: Dependencies due to human actions (testing, maintenance, diagnostic errors, incorrect calibration of sensors or instrument), which could affect manual actions or actuations of redundant systems, are reviewed in the system PSA analysis and modeled as separate events in the fault trees where necessary. Modeling of human action dependencies is discussed further in Section 15.4.
4. Intra-system dependencies: Intra-system dependencies (CCFs) can lead to multiple component unavailabilities from shared causes. The remainder of this PSAR section discusses the analysis approach for this type of dependency.

#### *Developing the Common Cause Failure Database*

Once the CCF basic events are identified, the model utilizes the CCF tool in CAFTA to automatically identify all the CCF combinations for a given CCF group and places the appropriate CCF BEs into the fault trees. CAFTA is also used to automatically quantify each of the CCF BEs.

As recommended by the CAFTA developers, common cause groups containing more than four components are treated simplistically. CCF events are developed for all combinations of one, two, and three components within the group, with all remaining failure probability assigned to a CCF event of all components failing. This is a conservative representation of the event modeling

all components failing, because the CCF probability of all components is overstated, but greatly simplifies the fault tree modeling.

When this treatment is too conservative, the calculations are completed outside of CAFTA and entered manually. In these cases, the CCF combinations are manually input to the fault trees using the parameters for the actual CCF group size and the CCF equations are directly input into the database file.

#### **15.6.1.3.9 Human Reliability Analysis**

The HRA is performed as part of the BWRX-300 PSA. Human actions play a key role in the risk assessment and may point to plant vulnerability early in the design phase that can be corrected before the design is finalized.

The HRA describes operator failure in a systematic and reproducible manner to facilitate a Human Error Probability (HEP) that is applied to the PSA model. The methodology is described below. Individual failure events generate reports that are presented in appendices to the PSA model.

#### **Scope**

The HRA covers three types of human actions:

- Those performed by maintenance or operations before an event
- Those performed by operations or maintenance to cause an event
- Those performed by operations after an event has occurred

Currently only post-initiator actions are documented given the limited plant design information.

#### **Inputs, Requirements, and Assumptions**

The primary input to the current HRA comes from a review of the success criteria and event sequences documents. These documents discuss the operator actions that are considered for the safe shutdown of the plant. The second resource is examination of the system fault trees to find actions that backup automatic functions. The third resource and the primary resource for pre-initiator actions comes from system engineers and any developed system maintenance schedules. Currently maintenance has not been considered in many of the systems and therefore pre-initiators are not included in the model.

#### **Requirements**

The requirements for the HRA analysis primarily come from the PRA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). Due to the PSAR design development stage and emergency procedures that are developed for the final safety analysis, this analysis is not expected to meet all the requirements of the standard at this time.

#### **Assumptions**

Generic assumptions used in the PSA:

1. All human actions are driven by procedure. No actions are considered that are outside operations documentation (e.g., actions based on knowledge, or mitigating actions outside the procedure).
2. Errors of commission are screened out from the analysis because they have negligible probability compared to errors of omission.
3. All persons performing operations or maintenance are adequately trained by an accredited training program.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

4. Post-initiator actions are all conducted from the control room unless otherwise specified in the human failure event report.
5. In the current phase of design, procedures for actions are based on engineering judgment and experience.
6. The pre-initiator HRA for a component that is made unavailable due to testing or maintenance, a restoration error is always possible.
7. The control room displays, and alarm annunciators are well-designed with critical alarms for all human failure events within the safety case identified and shown on the appropriate displays, with color-coding, lighting, and flashing to guide alarm management. Information is displayed in a manner appropriate to the type of display (e.g., large screen versus control desk versus local control panel, etc.).
8. Software-based display screens and control actuations are well-designed with consideration of human factors good practice principles including grouping, labeling, and spacing. In particular, potential selection errors either in screen navigation or in equipment icons on a particular screen are minimized through effective layout, screen item design, and group demarcations.
9. The time required to perform the steps in a task is estimated in most cases based on input from personnel that have BWR PSA human reliability analysis experience, have interacted with nuclear operators and is familiar with BWRs operator actions. Wherever possible, worst-case plant conditions and competing task workload are assumed ensuring conservative estimates. In some cases, the time taken to carry out task steps that are part of a task but are not necessary for successfully completing the specific action identified in the PSA are included in the overall task required time. This is done in cases where members of the HRA team indicate that these actions are likely to be attempted. Task timings are further examined and verified in later iterations of these analyses.
10. Assumptions related to individual dependencies include:
  - a. Although many actions occur over a long period of time, it is assumed that there are no intervening successes between actions
  - b. The same crew is assumed to perform the actions throughout the event
  - c. Cognitive demand is assumed different except where the actions are similar, e.g., alignment of cooling
  - d. Simultaneous actions are assumed when operators are working with multiple, immediate concerns (e.g., pump power and isolation)
  - e. The required resources to perform a task (operators, maintenance personnel, health physics, etc.) are estimated based on experience and engineering judgment
11. Timing overrides are implemented as necessary to place actions into chronological order. In the current model, no overrides are necessary.

## **HRA Discussion**

There are several types of human actions during normal plant operation:

- Those related with the failure to restore equipment to its normal condition following a test or maintenance action
- Those causing an IE

Human actions during an accident are:

- Those performed by operators following established procedures for mitigating the consequences of an accident
- Those performed in the mistaken belief that they are the appropriate actions as indicated in the procedures, but in reality, can worsen the conditions of the accident, thereby complicating the mitigation process
- Those that are not explicitly included in the procedures and are used to recover the operability of failed equipment or use alternative means

Treatment of human actions in the PSA leads key to realistic understanding of event sequences and their relative importance to overall risk. The HRA is performed in a structured framework that allows development of the analysis in an orderly fashion while considering the general boundary of the PSA.

There are three main types of human actions in the HRA:

*Type A: Pre-Initiating Event Human Actions*

Before an IE, plant personnel affect availability of standby systems by inadvertently disabling equipment or failing to restore the correct position during the performance of operational activities in the plant. Example: maintenance staff incorrectly perform a valve lineup after maintenance on a pump. When the pump is needed during an event, it cannot run or is damaged.

*Type B: Initiating Event Induced by Human Actions*

Plant personnel initiate an event through interactions with plant equipment. These typically occur due to a mispositioning or incorrect operation of equipment that inadvertently trips equipment or inserts false control signals. Example: One of three pumps providing service water to a condenser fails and operators do not align a backup in a timely manner. The plant trips on loss of service water.

*Type C: Post-Initiating Event Human Actions*

After an initiating event has occurred, plant operations attempt to place the plant in a safe, stable state using emergency operating procedures. There are two types of errors associated with Type C actions:

1. Commission Errors – By following procedures during the course of an accident, plant personnel operate standby equipment that aggravates the accident. This action, called a commission error, gives rise to a situation in which the operators place the plant in a worse condition than if they had done nothing (i.e., an omission error). The risk of commission errors is considered insignificant when the plant has EOPs that have a symptom-based orientation and the EOPs cover all the possible scenarios in the event sequences analysis.
2. Omission Errors – Failure to perform a step in the procedure, or failure to act in a timely manner results in either equipment damage or core damage. It is imperative that operations perform the correct steps within the time frame before damage occurs.

**Review of Plant Practices**

Once the pre-accident errors are identified as part of the systems analysis, the human reliability analyst examines the pertinent procedures to determine which critical components are placed in off-normal configurations and if not restored to their normal alignment, could render the component, train, or system inoperable. Examples of errors that render components, trains, or system inoperable include:

- Prior to performing maintenance on a pump, the protective tagging order typically involves isolating the pump by closing both the suction and discharge isolation valves, racking out the pump breaker, and placing the pump control switch in pull-to-lock. Failure to return these components to their normal position will result in unavailability of the pump.
- In the case of testing, failure to close the test line isolation valve can result in unacceptable degraded performance of the train or system.
- During instrument calibrations, a I&C technician errantly calibrates an instrument that automatically actuates, automatically trips, or causes some other actuation.
- While performing maintenance on a transmitter, the instrument isolation valves are left closed resulting in erroneous operation or failure to detect system parameter changes such as pressure or level changes.

Plant practices for operating BWRs are used until plant-specific maintenance procedures are developed.

#### **15.6.1.3.10 Event Sequence Frequency Quantification**

The BWRX-300 PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The calculation of core/fuel damage and release category frequencies is performed as single gate for each sequence and release category. The top gates include all sequences but uses a sequence marker to identify the event tree sequences in the cutsets generated by the single gate. The use of a sequence marker results in retaining all the minimal cutsets for that specific sequence but prevents subsuming of non-minimal cutsets among sequences. This is addressed later in the recovery process. The contribution to risk from these non-minimal sequences is generally small, although this can affect the results for specific hazards, such as earthquakes (seismic).

The sequence logic is set up to exclude any combinations associated with the success branches in the specific sequence. The individual sequence results are then combined for reporting, analyzing, or used as input for the L2 or L3 portions of the PSA.

#### **Methodology**

The purpose of the event sequence frequency quantification is to obtain the Boolean equation corresponding to the radionuclide release. The quantification is developed in terms of minimal cutsets representing the minimal combinations of events that result in radionuclide release.

The following key aspects characterize the release category frequency quantification process:

- Event trees model plant response to each group of IE
- Fault trees model the behaviour of front-line and support systems
- Integration of event tree and fault tree structures into a single linked model
- Quantification of the linked Boolean model with the probabilistic database and boundary condition files (flag files)

#### ***Use of Computer Aided Fault Tree Analysis for Solving the PSA***

The EPRI Integrated Risk Technologies User Group offers the suite of risk software tools (formerly known as R&R Workstation) that are used for the Level 1 PSA model development and quantification. These risk software tools include CAFTA for event tree and fault tree construction and PRAQUANT for master model integration and quantification. Accident sequence frequencies

are derived from the quantification results of a single top master fault tree by post-processing using the sequence markers.

The FTREX code is used in generating Minimal Cutsets (MCS) from fault tree logic models. FTREX uses a zero-suppressed binary decision diagram method for interpreting large fault tree models when developing MCSs. FTREX is built for PSA applications and is applicable to the BWRX-300 PSA without specific limitations.

Chapter 3, Appendix 3G discusses the computer programs used in the PSA.

The reliability databases supporting the BWRX-300 PSA include the IE frequencies, component failure frequencies and probabilities, CCF data, component repair times, test intervals and durations, mission times, maintenance unavailability, unavailability due to testing, and human error data. Uncertainties for data values are also included. All the data sources are justified and documented.

#### *Additional Quantification Steps*

The quantification process used for the PSA is iterative and includes numerous stages where internal and external review is performed before final results are documented. Key steps performed during the quantification include:

1. Circular Logic: It is possible to generate a master fault tree that has circular logic. This is where a gate is used as input to a second gate that happens to also be an input somewhere in the tree of the first gate.
2. Basic Event Consistency Check: The IE frequencies and the probabilities associated with BEs of the model are consistent with the definitions of the events in the context of the logic model. As the PSA cutsets are reviewed, they are reviewed for consistency in terms of the applied assumptions, specific design, and operational experience.
3. Mutually Exclusive Event Review: The model is solved to verify the event combinations are removed.
4. Human Error Dependencies: One example of how human failure error dependencies are determined is summarized:
  - a. All post-initiator HFE probabilities are set to one (or a high value).
  - b. The single top quantification is performed again. The most critical human failure error dependencies align, and this is apparent in the cutset results.
  - c. The PSA then uses the results generated in the cutsets to generate recovery rules that change the probability of these joint failures to a value that better reflects the dependency between these human failure errors taken from the HRA.
5. Recovery Actions: For a design level PSA, equipment recovery is not credited other than operation of backup systems that is addressed in the Type C operator actions. In the site-specific PSA, recovery is possible for longer term accident sequences where equipment recovery is possible.
6. Truncation Justification: To justify an appropriate truncation value is used, a test for CDF convergence is performed. Additionally, the last two convergence runs are used to develop a list of risk-significant accident sequences (e.g., contributing greater than 1%) to ensure no new significant accident sequences are identified by lowering the truncation limit further.

7. Importance Analysis: This analysis documents the overall contribution of basic events to each phase of the PSA including the Level 1, Level 2, and Level 3 PSA. Separate importance results are provided for basic event types such as IE, human errors, event class, release categories, etc. Together with the uncertainty and sensitivity analyses, this satisfies CNSC REGDOC-2.4.2, Requirement 4.10 (Reference 15.6-1).
8. Uncertainty and Sensitivity
9. Review of the results: The U.S. PSA Standard and the IAEA-TECDOC-1804 (Reference 15.6-13) require the review of the results in detail to ensure the results are consistent and justified. For the BWRX-300, the comparison of these results with similar plants is not possible. As a result, a general comparison with the ESBWR or other BWRs using similar components (e.g., ICS) is made.

#### **15.6.1.4 Internal Fire Hazards**

The probabilistic fire analysis was performed with simplifying assumptions because the specifics of cable routings, ignition sources, or target locations in each zone of the plant are still in the design phase. Because of this limitation, a simplified, conservative, and bounding approach is used in this analysis.

The current scope of the analysis is for at-power accident scenarios.

#### **Methodology**

The BWRX-300 internal Fire PRA is performed according to the guidance in USNRC NUREG/CR-6850 (Reference 15.6-20). Other key references may be used:

- USNRC NUREG/CR-6850, EPRI 1019259, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements" (Reference 15.6-27)
- USNRC NUREG-2169, EPRI 3002002936, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database" (Reference 15.6-28)
- USNRC NUREG/CR-7150 Vol 1, BNL-NUREG-98204-2012, EPRI 1026424, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," (Reference 15.6-29)
- USNRC NUREG/CR-7150 Vol 2, BNL-NUREG-98204-2012, EPRI 3002001989, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Fire," (Reference 15.6-30)

USNRC NUREG/CR-6850, EPRI 1011989 (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1, EPRI 1019259 (Reference 15.6-27), document state-of-the-art methods, tools, and data for the conduct of a Fire PRA for a commercial nuclear power plant application. The methods have been developed under the fire risk re-quantification study. This study was conducted as a joint activity between EPRI and the U.S. NRC Office of Nuclear Regulatory Research under the terms of an NRC/EPRI memorandum of understanding and an accompanying fire research addendum.

For the BWRX-300 Fire PRA model development, the following USNRC NUREG/CR-6850 (Reference 15.6-20) tasks are applicable:

- Task 1: Plant Boundary & Partitioning



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- Task 2: Fire PRA Component Selection
- Task 3: Fire PRA Cable Selection
- Task 4: Qualitative Screening
- Task 5: Fire-Induced Risk Model
- Task 6: Fire Ignition Frequencies
- Task 7: Quantitative Screening
- Task 12: Post-Fire HRA
- Task 14: Fire Risk Quantification
- Task 15: Uncertainty and Sensitivity Analyses
- Task 16: Fire PSA Documentation

Some of the above tasks are simplified while others are omitted at this stage of the plant design. Tasks not addressed in this study include:

- Scoping Fire Modeling (Task 8)
- Detailed Circuit Failure Analysis (Task 9)
- Circuit Failure Mode Likelihood (Task 10)
- Detailed Fire Modeling (Task 11) including multi-compartment analysis
- Seismic Fire Interaction (Task 13)
- Support Task A Plant Walk Downs is not performed in the design phase

The Fire PRA database is created with an Access®-driven database (FRANX) that includes all tables that are necessary to develop a scoping-level Fire PRA model. Enhancements to the Fire PRA database are possible in future updates once more details are available for cable tray routes and their contents as well as more detailed ignition source locations.

Fire ignition frequencies for power operation at each plant area (physical analysis unit) are estimated using the USNRC NUREG/CR-6850 (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1 methodology (Reference 15.6-27) and data from USNRC NUREG-2169 (Reference 15.6-28). Fire frequencies for shutdown conditions are not developed at this time.

For a postulated fire, a list of impacted components is generated with the mapping defined in the Fire PRA database. A list of impacted components is also generated with the assumed cable routing. The cable routing is assumed based on the modeled PRA components, their supports, and the general room layout of the BWRX-300 design. Fires are conservatively assumed to propagate unmitigated in each fire area (no suppression is credited) and damage all functions in the fire area. The internal events PRA accident sequence structures and system fault trees and success criteria are used in the calculation of the fire CDF.

The BWRX-300 Fire PSA employs the following EPRI software:

- Fault Tree Reliability Evaluation eXpert. This software generates cutsets from the fault trees produced in CAFTA.
- CAFTA. This software is used to build the logic model of the plant, producing all the fault trees and event trees.

## Assumptions

1. A fire ignition in any fire area may grow into a fully developed fire.
2. The analysis does not take credit for any fire suppression (i.e., self-extinguishment, installed suppression systems, nor manual firefighting activities). All fires disable all potentially affected equipment in the area in the analysis.
3. Unless otherwise stated, a fire causes failures of all fire-susceptible components in the subject fire area and detailed fire modeling is not performed in this revision of the BWRX-300 Fire PRA.
4. Recovery of the failed components or cables after the postulated fire is not credited.
5. Unless otherwise stated, the analysis does not credit the distance between fire sources and targets.
6. All fire-induced equipment damage occurs at time zero in the scenario progression.
7. Cable routing is postulated for PRA purposes using plant general arrangement drawings with major electrical component locations because cable routing is provided later in the design. A list of cables is generated that includes all modeled supports for PRA components included in the current PRA model. This list captures most cables, especially for expected risk-significant components.
8. Fires in the MCR and SCR are currently modeled as impacting components associated with any assumed cable routings that go through the rooms but do not terminate in the MCR or SCR (i.e., the assumed cable route for a bus that goes through the MCR between the TB and the B train MCC room in the CB). The MCR and SCR are not assumed to impact components where control or visualization cables may be in the rooms. This is a realistic treatment because plant design could be similar to the ESBWR where the control room controls are connected to the Distributed Control and Information System (DCIS) rooms (which would be unaffected by a MCR fire), via fiber cables. The loss, including melting, of the fibers or visual display units (VDUs) will not cause inadvertent actuations, nor affect the automatic actuations associated with safety and non-safety equipment. In addition, fires in the MCR and SCR are assumed to fail all modeled human failure events.
9. All finalized details of cable route information are not yet available. Individual routes are assumed based on location of PRA modeled power/signal sources and the component.
10. The cable routings use general separation criteria because of the number of different signal and power divisions where there are physical analysis units with more than one division and power and control cables postulated. As the BWRX-300 design matures, cable routings are refined to include any credited fire barrier(s) within an overall physical analysis unit that serve to separate divisions.
11. Fire ignition frequencies remain constant over time and are based on industry generic fire frequencies. At operating plants, the total ignition frequency is the same among plants for the same equipment type, regardless of differences in the quantity and characteristics of the equipment type that exists among the plants. This is conservative because the BWRX-300 design has significantly fewer pumps, motors, and other active components than earlier plant designs upon which the current plant generic ignition frequencies are based.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

12. All ignition source type bins are applicable to the BWRX-300 plant with the following exceptions:
  - a. Bins 02 and 03 are not applicable since they are used for pressurized water reactor plant designs.
  - b. Bin 22 for scram motor generator sets is not applicable to BWRX-300.
  - c. Bin 04 is not applicable to the BWRX-300 MCR or SCR because the BWRX-300 design is completely digital as opposed to traditional electro-mechanical designs. However, to ensure a conservative analysis is produced, twenty-five percent (25%) of the traditional Bin 04 ignition frequency is assumed to be applicable and is assigned to the BWRX-300 MCR ignition frequency.
13. Although the BWRX-300 plant may be located with existing NPPs, the plant location weighting factors are one because the BWRX-300 plant is designed as a single-unit plant with no shared buildings.
14. Because the BWRX-300 plant is in the preliminary design phase, the count of components is performed with the modeled PRA components as well as preliminary design layout drawings.
15. Because the plant has not operated and no history of maintenance activity is available, the weighting factor evaluation is simplified. It is assumed that all compartments (physical analysis units ) have the same transient fire influencing factors with a value of one is used for all influencing factors.
16. All fires result in a manual reactor shutdown regardless of location or potential induced failures.
17. Full details of electrical cabinet locations are not yet available. Preliminary design information notes the division or train of equipment is used in the development of this revision of the BWRX-300 Fire PRA for electrical cabinet and high energy arching fault counts.
18. The containment is always inerted during power operation. The possibility exists that the containment could be de-inerted in preparation for an outage, or still de-inerted when coming out of an outage. Typically, de-inerting occurs around 2-3 days before an outage that can result in a fire during this time. This time window is not included in the current BWRX-300 Fire PRA but is included in the future shutdown operation BWRX-300 Fire PRA.
19. Cables associated with display or visualization component status or function may be routed between a component or instrumentation location and the MCR or SCR are not impacted by any automatic function of the component.

### **Modeling for Full-Power Condition**

Detailed fire modeling is not performed for this revision of the BWRX-300 Fire PRA for at-power operation. Detailed fire modeling is undertaken when the BWRX-300 design is more mature and additional details on equipment locations and cable tray routes are known with certainty.

For this revision of the BWRX-300 Fire PRA, full room (physical analysis unit) burnout scenarios are used where any PRA-related component and cable within the particular physical analysis unit is failed at the total physical analysis unit ignition frequency during Fire PRA quantification.

### **Evaluation of Multi-Component Scenarios**

Fire propagation cases (multi-compartment scenarios) from one physical analysis unit into another physical analysis unit are not currently postulated for the fire areas listed in the at-power Fire PRA. As the BWRX-300 design matures, this analysis is undertaken.

### **Evaluation of Potential Smoke Damage**

Detailed HVAC system design information is not yet available for the BWRX-300 plant design. Consequential failures from potential smoke damage cannot be assessed. This is provided in future updates to the BWRX-300 Fire PRA as the plant design matures.

### **Detailed Fire Modeling for Shutdown Condition**

The shutdown PRA model is not yet constructed but is expected to be based on the Level 1 internal events PRA model. Detailed fire modeling follows the same process described previously with the exception that the evaluation of the applicability of shutdown conditions in the Fire PRA model is provided in future updates to the BWRX-300 Fire PRA.

#### **15.6.1.5 Internal Flooding Hazard**

The BWRX-300 internal probabilistic flood analysis objective is identifying and providing a quantitative assessment of the radionuclide release risk due to internal flood events. It models potential flood vulnerabilities in conjunction with random failures modeled as part of the internal events PSA. Through this process, flood vulnerabilities that could jeopardize core integrity and containment integrity are identified.

The floods may be caused by large leaks due to rupture or cracking of pipes, piping components, or water/fluid containers such as storage tanks. Another possible flooding cause is the operation of fire protection equipment.

Excluded from this analysis is flooding associated from external sources that is considered under the External Flooding Hazard Analysis for localized flooding events and intense weather events. These external flooding events are reasonably precluded from the BWRX-300 probabilistic flood analysis because the standard plant design prevents external flooding conditions affecting the safe operation of the plant.

### **Methodology**

The internal PSA flooding analysis identifies and classifies potential flooding sources and events. Component location and data is compiled generating a frequency of occurrence that represents the effects associated with each of the potential flooding events. In addition, an evaluation is performed to identify, screen, and quantify specific plant effects/failures associated with each flooding event. Finally, the BWRX-300-specific flooding frequencies and plant effects are applied to the PSA model to obtain risk results. For the BWRX-300 Flooding PSA model development, the following tasks are performed:

- Identification of flood sources
- Development of flooding scenarios
- Development of flooding frequencies
- Analysis of flooding scenarios

The internal PSA flooding analysis is based on the design basis BWRX-300 SSCs. Critical to the flooding analysis is the location of these features and their interaction with other BWRX-300 SSCs. The current list of system components and location of equipment is based on the current

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

design and plant layout drawings and relies on the component location analysis collected for the Fire PSA step (e.g., common spatial information).

The development of BWRX-300-specific flooding scenarios requires a detailed analysis of data including plant component location, system capacity, and potential failure mechanisms. Following the identification of potential flooding scenarios, characteristic scenarios are selected as representative of flood areas and subject to quantitative analysis.

Data is collected from industry sources for the BWRX-300 equipment and system components. Failure data for consideration in the flooding analysis includes piping runs, pumps, valves, tanks, heat exchangers, and circulating water expansion joints to develop the severity and effect of potential flooding scenarios. These failure rates in combination with types and capacity of system components located within specific flood zones are used to develop the flooding frequencies and frequency uncertainties. Flooding frequencies for both large break and small leak scenarios are developed for each flooding scenario.

The flood scenarios for each flood area are quantified to calculate a probabilistic risk value and summed to provide an overall risk analysis. USNRC NUREG/CR-4639 and NSAC-60 provide additional guidance for flooding analysis. The requirements for a flooding PSA are discussed in Section 3 of the PSA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

Walkdowns are a critical task for a mature flooding PSA and is performed and documented after the plant construction is complete.

#### *Key Inputs and Assumptions*

The following are inputs for the BWRX-300 flooding analysis:

- At-power internal events PSA model
- BWRX-300 design features for protection against flooding
- Flood zone spatial information

The following assumptions are used in support of the BWRX-300 flooding analysis:

1. Flooding resulting from component ruptures
2. For each tank rupture, the entire tank inventory is drained
3. Non-qualified submerged equipment (motors or solenoids for valves, control cabinets and circuitry) fails if the water level in the flood zone reaches a level of 1 foot above floor elevation
4. If equipment fails, the equipment fails at the start of the flood
5. The expected effect of flooding electrical equipment such as motor control centers, electrical cabinets, and terminal boxes, is a short to ground, removing power from the loads served by the component (all electrical equipment failures are treated as ground shorts)
6. Flooding and/or spraying of a motor-operated valves (MOVs) causes the valve to fail as-is
7. Passive components, such as pipes and tanks are not considered vulnerable to flooding effects
8. Concrete walls are considered flood barriers, capable of withstanding the expected maximum flood loading, and remain intact throughout a flooding event

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

9. Electrical circuit fault protection provides protection for plant electric circuits via protective relaying, circuit breakers, and fuses and does not result in the loss of the bus that supplies power to the affected component
10. For floor drains, appropriate precautions such as check valves, back flow preventers, and siphon breaks prevent back flow and any potential flooding into connected plant areas (propagation of flooding through the drain system via failed check valves is not considered)
11. Doors in electrical equipment rooms within the RB are evaluated for watertight doors for flooding up to the ground level elevation. For the flooding analysis, the watertight doors are normally closed at-power and opening of these doors generates an alarm in the control room with procedures that direct their immediate closure upon alarm receipt
12. Dry pipe systems (such as a pre-action Fire Protection System) with closed sprinkler headers are not modeled as flood sources due to the low frequency failure of the dry pipe coincident with spurious opening of the actuation valve
13. Equipment located in the yard is not considered susceptible to internal flooding damage
14. Human-induced mechanisms such as overfilling tanks or diversion of flow created by maintenance are not modeled because operating and maintenance procedures have not been developed, and frequency of maintenance and duration of maintenance have not been determined. They are included in the PSA updates when these inputs are available
15. The flood volume retained by the drain system is not credited in the flooding analysis for large flooding scenarios. Therefore, the capacity of the drain system has not been estimated
16. The amount of water retained by sumps, berms, dikes, and curbs is negligible. This is conservative because retained water provides additional time for flooding response
17. Flooding initiator frequencies remain constant over time
18. Among the plant buildings, flooding frequency is the same for the same equipment type, regardless of difference in the quantity and characteristics of the equipment type that may exist among the plant

#### **15.6.1.6 High Wind Hazard**

The BWRX-300 high wind analysis quantifies event sequences and containment releases initiated by both tornado and straight winds. Straight winds are lesser velocity winds that pose minimal challenges to the plant design. Straight-line winds of lesser velocities are not included in the BWRX-300 high wind risk analysis. The consequences of these winds is addressed as part of the weather-related LOPP IE in the at-power internal events PSA model.

#### **Methodology**

The high wind risk analysis involves the following major steps:

- Tornado high wind event frequency
- Tornado-induced plant effects
- Analysis of tornado-induced release category frequencies
- Analysis of straight-line wind effects

The high wind PSA analysis identifies and classifies potential high wind events. Data is then compiled to generate an occurrence frequency that represents each potential high wind events.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

In addition, a qualitative evaluation of the design basis BWRX-300 SSCs is performed identifying specific plant effects/failures associated with each high wind event. A BWRX-300-specific high wind event frequency and plant effects are applied to the at-power model to obtain risk results. High wind hazards are characterized by their impacts (e.g., dynamic load from gusts, averaged loading, rotation velocity, pressure differential, tornado path, missile impact potential).

The BWRX-300 is designed for a tornado wind load that is the maximum wind speed such that it does not challenge safety structures.

ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12), including the guidance in the non-mandatory appendix, provides requirements and guidance on performing detailed high winds PSA. The changes in Part 7 of ASME/ANS PSA Standard are the most significant changes and reflect advancements in high winds PSA. Included in the standard requirements is the consideration of correlated hazards and hazard effects, such as the potential for local flooding associated with high winds or the impact of high wind-driven rain. This standard is utilized for both the screening of potential hazards and the analysis of any unscreened hazards.

The BWRX-300 High Wind PSA uses the following EPRI software:

1. Fault Tree Reliability Evaluation eXpert - this software generates cutsets from the fault trees produced in CAFTA.
2. CAFTA- this software is used to build the logic model of the plant, producing all the fault trees and event trees.
3. ACUBE 2.0 -this software provides a more accurate solution to the cutsets than conventional solutions.

### **Key Inputs**

Site-specific data are inputs for the external hazards PSA analyses. Site-specific wind hazard analysis generally requires the use of regional data. The size of the region requires judgment and depends on the regional climatology and type of wind hazard, the number of years accurate records are available, the extent and quality of the data, and the hazard's spatial variability within the region. In the design phase, it is possible to select a bounding region for a high wind hazard; however, a bounding region for tornadoes may not be the bounding region for hurricanes.

The high winds equipment list is evaluated based on the internal events PSA required functions and supporting SSCs. The internal events PSA is used to quantify the high winds PSA with modification of the damaged or potentially damaged SSCs.

### **Key Assumptions**

The following assumptions were used in conducting the BWRX-300 high wind risk analysis:

1. The winds classification used in the BWRX-300 high wind analysis is based on the Saffir-Simpson scale.
2. The tornado winds classification for the BWRX-300 high wind analysis is based on the recommended wind speed ranges of the Enhanced Fujita scale (EF-scale).
3. All at-power BWRX-300 high wind analyses include high winds assuming the plant is operating. This approach is conservative for the high wind analysis where sufficient advanced warning and procedures allow the plant to be placed into a safe condition (shutdown operations) prior to the high wind event.

4. For the BWRX-300 high wind analysis, the frequency of occurrence data for tornadoes high wind events is compiled from sources based on the Enhanced Fujita scale (EF-scale) for classification of tornadoes. This data is correlated to the BWRX-300 EF-scale of tornado classification.

#### **15.6.1.7 Seismic Hazard**

The scoping Seismic PSA (SPSA) for the BWRX-300 Standard Plant Design describes the methodology used:

- Presents the seismic hazard curve used
- Lists the generic structure and component fragilities and sources
- Presents results by ground motion level and important fragilities
- Documents seismic risk insights
- Identifies plant design requirements

The model used is a preliminary scoping model that provides important insights into plant design. The SPSA activities include:

- Identifying IEs applicable to seismic events
- Developing a seismic equipment list
- Assigning representative fragilities applicable to SSCs
- Adjusting for human error probabilities to reflect increased distractions and stress from the seismic events
- Modeling seismic Level 2 PSA in the updated SPSA

The scoping SPSA is a full-power model. Seismic-induced flooding and fire, contact chatter, building-to-building interactions, and soil liquefaction are addressed qualitatively in the scoping stage.

Although for low ground motion levels offsite power may be available, offsite power is assumed unavailable in all modeled accident sequences.

The plant operational and emergency systems are represented by event and fault trees. The analysis gives insights into the dominant contributors to seismic risk.

#### **Methodology**

The BWRX-300 scoping SPSA development involves four major activities:

1. Probabilistic seismic hazard analysis
2. Seismic fragility analysis
3. Seismic plant response analysis
4. SPSA quantification

The probabilistic seismic hazard analysis identifies ground motion bin IE frequencies that are based on the seismic hazard curve for the Darlington site. This curve is then partitioned into several ground motion bins.

The BWRX-300 SPSA results are reported based on truncating the seismic hazard at the 1E-6/yr earthquake frequency, consistent with IAEA-TECDOC-1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants" (Reference 15.6-21).



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

For some external hazards, the IAEA Safety Requirements states that “it may not be practical or even possible to demonstrate that the occurrence of a hazard of such severity could cause extensive plant damage leading to a large or early radioactive release and therefore needing to be practically eliminated, is below a threshold of frequency such as  $10^{-6}$ /year”. Seismic events with exceedance frequencies less than  $1\text{E-}6/\text{yr}$  are attended by significant uncertainty. The IAEA guideline is applied to the SPSA results by truncating the seismic hazard. Applying the seismic hazard truncation at the  $1\text{E-}6/\text{yr}$  earthquake frequency results in the sum of the CDFs for seven ground motion bins.

The seismic fragility analysis produces a seismic equipment list that includes a list of plant SSCs related to seismic-induced IEs and SSCs that mitigate plant responses to IEs. Fragilities for the SSCs on the seismic equipment list are selected in the seismic fragility analysis.

The SPSA model is developed from the seismic plant response analysis. The seismic plant response analysis uses a seismic IE tree to model the plant structures that are required to remain intact and the probability of a seismically-induced LOCA, main feedwater line break and main steam line break. A LOPP is assumed to occur for all SPSA modeled sequences. The end states of this event tree that do not go directly to core damage are linked to the appropriate internal event trees.

An updated internal events top logic model is used as input to the SPSA top logic model development update. This updated internal events top logic model is used to update the systems fault trees and PSA event trees reflecting the current evolution of the BWRX-300 design. The internal events top logic addresses:

- Updated system fault trees to reflect current system designs
- Updated Level 1 event trees
- Updated model continues to credit containment vent for decay heat removal
- Updated model continues to credit CRD injection, to control inventory during failure to scram and LOCAs
- Updated model continues to credit ultimate pressure relief for transients, small LOCA and failure to scram

For the SPSA quantification, the FRANX code is used to integrate the seismic hazard with the SSC fragilities to produce the SPSA CDF results.

The BWRX-300 SPSA uses the following EPRI software:

- FRANX uses the internal events model and adds seismic initiators and hazard curves that results in the CDF for each ground motion level
- Fault Tree Reliability Evaluation eXpert generates cutsets from the fault trees produced in CAFTA
- CAFTA is used to build the logic model of the plant, producing all the fault trees and event trees
- ACUBE 2.0 provides a more accurate solution to the cutsets than conventional solutions

## Assumptions

The following principal assumptions are used in developing the seismic PSAs:

1. A LOPP is modeled to occur for all seismic accident sequences. Although offsite power may be available for low ground motion level earthquakes, this assumption is made to simplify the model. A consequence of this assumption is that power conversion system is unavailable. This assumption is made in this scoping model, but in the final model, the offsite power system is modeled explicitly.
2. The seismic-induced failures of the RB, reactor support structure or RPV are assumed to result in core damage and containment bypass.
3. Fragility group MV-MSIV consists of the MSIVs and RIVs. It is conservatively assumed in this assessment that the two sets of valves are similar in design response to be seismically correlated, meaning that if one fails, the other will fail as well. This assumption is revisited in the final model using the methodology discussed in USNRC NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)" (Reference 15.6-22), these valves have uncorrelated fragilities due to dissimilarities. The outboard MSIV and MSRIVs are treated as fully correlated and with further analysis may be treated as partially correlated.
4. Chatter of electrical contacts, such as relay contacts, is not addressed. Contact chatter is not expected to contribute significantly to BWRX-300 seismic risk. The DL4a ICS actuation and the DL4 scram actuation are expected to fail-safe as a result of chatter and that do not include seal in circuits that would interfere with actuation. Contact chatter assessment is addressed in a future revision of the SPSA.
5. The SPSA utilizes fragilities primarily from the ESBWR. Where ESBWR data are not available, representative fragilities from EPRI Report 3002000709, "Seismic Probabilistic Risk Assessment Implementation Guide" (Reference 15.6-23) are used. For risk-significant fragility groups, representative fragilities developed for the BWRX-300 are used.
6. The electrical equipment that supports DL2 and DL3 actuation signals is seismically correlated and is modeled by a fragility group. This modeling is conservative because the equipment is not collocated. DL2 electrical equipment is located in the control building and the DL3 electrical equipment is located in the RB. Although DL3 is designed to fail in the fail-safe condition, DL3 ultimately relies upon digital electrical equipment to function and DL3 actuation is assigned to a fragility group.
7. In this scoping model, the fractions of the small LOCA frequency that occur in the CUW and the CRD systems are the same for seismic events as for random failures. This simplification is compared to plant-specific data in the final model.
8. Dependencies of multiple human failure events in the same cutset are not modeled. Because there are few human failure events in the model, human actions are not as important in the BWRX-300 as they are in most operating plants, and the nominal HEPs are screening values where dependency is not likely to be important.
9. The diesel generators located in the TB are air-cooled. The seismic failure of the TB is modeled to fail both diesel generators. The seismic failure of the intake structure does not impact the diesel generators.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

10. At this stage of the analysis, equipment interaction concerns (e.g., equipment falling on other equipment) are not evaluated. As equipment location is finalized, equipment interaction is addressed.
11. All redundant components in the same system are correlated. This is reasonable for components that are identical, of the same orientation, and on the same floor. The MSIVs and the MSRIVs correlation may be partially correlated as the design evolves. As more detailed designs are developed, other instances of partial correlation may be found and modeled using the USNRC NUREG/CR-7237 methodology (Reference 15.6-22) and re-evaluated in the final model.
12. Generic fragilities are used in the preliminary SPSA. This is sufficient for the scoping study because the fragilities are not site-specific.
13. The DNGS seismic hazard curve is used as input to the BWRX-300 SPSA. In the final model, a site-specific hazard curve is used.
14. A break in the CUW piping could result in a break outside containment if the RWB that is not a safety structure (it is a hardened structure that meets OBE or  $\frac{1}{2}$  SSE earthquake) is damaged or the CUW heat exchangers are damaged. This scenario is evaluated when the design is finalized.
15. The polar crane is designed, or its position administratively controlled by plant procedure during normal operations so that seismic damage of the crane does not impact safety class RB contents, such as the ICS pools and heat exchangers.
16. Equipment anchorages comport with the assigned fragilities, and this is verified as the design progresses.
17. The existing DNGS site has no impact on the SPSA for the BWRX-300 and is a single site.
18. In this current scoping model, equipment screened out of the internal events PSA due to low failure likelihoods, such as ductwork, dampers, piping, and cable trays, are not added and are included in the fragilities that are modeled. In the final model, as more plant-specific fragilities are developed, these components are added.
19. If a building fails, all equipment in that building also fails.
20. If the CB fails, all human failure errors except the human failure errors to align and operate diverse and flexible mitigation strategies (FLEX) fails. The final model addresses whether the CB failure also fails FLEX due to unavailability of operators or lack of a path to the FLEX equipment. Since the HEP for aligning FLEX is 1.0 at the most risk-significant ground motion levels, allowing FLEX with CB failure is of little consequence.
21. A failure of the TB fails both steam lines and feedwater lines because these lines extend into the TB.
22. The FLEX system allows water from Lake Ontario pumped to the reactor, fuel pool, or ICS.
23. Unisolable LOCA is not modeled by the SPSA at this time.
24. Fragility group CR-RODS, Failure of Control Rods to Insert is completely correlated. If control rod scram function is damaged, all control rods are assumed to fail to fully insert.
25. For RPV rupture (fragility group CONRPV), damage occurs in the lower portion of the RPV.

As part of the scoping Seismic PSA, a sensitivity analysis is performed to identify the safety-significant seismic fragility groups. This sensitivity analysis identifies the BWRX-300 SSCs that contribute significantly to the seismic safety profile.

Nine fragility groups are identified by the sensitivity analysis to have seismic safety significance. Of these nine, one group (the reactor building polar crane) is expected to be mitigated by administrative controls rather than seismic design/analysis refinement. The other eight fragility groups are:

- Control Rods
- RIVs/MSIV
- IC HX
- ICS Actuation Valves
- RB
- RPV
- RPV Support Structure
- Scram HCUs

As the detailed design progresses, updated seismic capacities are developed. These seismic capacities are then fed back into the seismic PSA and new risk metrics are recalculated in an iterative process. This risk-informed design process ensures that the DNNP BWRX-300 safety goals are met.

#### **15.6.1.8 Level 1 PSA – Low Power and Shutdown Risks**

The Low Power Shutdown (LPSD) Risk PSA is developed following the requirements of the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). Currently, the ASME/ANS standard for LPSD is issued for trial use and pilot application. After the trial use, feedback was provided to the PSA Standard committee and the LPSD standard is being revised.

A detailed PSA is performed to determine the radionuclide release risk during shutdown. Loss of CUW, SDC, Plant Cooling Water System (PCW) and LOPP are all evaluated for radionuclide risk during shutdown. The approach for the LPSD PSA is similar to the Full-Power PSA, involving fault trees and event trees used in determining the shutdown risk for each IE analyzed.

The evaluation encompasses plant operation in hot shutdown, cold shutdown, and refueling modes (Modes 3, 4, 5, 6) while Modes 1, 2, 3, 4 are bounded by the at-power PSA model. During these modes, the plant is transitioned through several POSs that are distinguished in the LPSD PSA by different plant conditions and configurations. The LPSD PSA addresses POSs where there is fuel in the reactor vessel. It includes all aspects of the NSSS, the containment, and all systems that support operation of the NSSS and containment.

The event sequences in the Shutdown PSA are classified according to whether the core is damaged. All shutdown core damage sequences lead directly to a radionuclides release to the environment (containment is assumed to be open at the time of the IE). In the final PSA, this assumption is revisited to accurately assess the PSA results.

The critical safety functions credited in the shutdown model are decay heat removal and inventory control. The containment function is credited for POSs where containment is not open. Because the containment is open for many of the POSs, containment integrity is not relevant to any modeled functions. Reactivity control is assumed to have no significant effect on the shutdown

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

model. Electrical power availability, as well as the availability of other support systems, are modeled for the effect on decay heat removal. LOPP is evaluated as an IE, and power dependencies for systems are included in the model in a similar manner to the at-power PSA.

The LPSD PSA includes fuel and heavy load movements where the PSA analyzes heavy load drops that can cause fuel damage, core damage, or large releases. The PSA results for heavy load drops considers the following events:

- Fuel pool leak due to rupture At-Power
- RPV leak due to rupture (below top of fuel) during Low Power and Shutdown
- RPV leak due to rupture (at feedwater nozzle) during Low Power and Shutdown
- Fuel Pool leak due to rupture during Low Power and Shutdown
- Loss of ICS Train A or SDC during Low Power and Shutdown
- Loss of ICS Train B or SDC during Low Power and Shutdown

### **Methodology**

ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) provides input guidance documents for the Low Power and Shutdown PSA model development, including the LPSD PSA Standard, and IAEA-TECDOC-1144, "Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes" (Reference 15.6-24).

### **Plant Configurations in Low Power and Shutdown Conditions**

Differences between the Low Power and Shutdown PSA and the Power Operation PSA are attributed to:

- Plant operating mode
- Plant operating state including configuration
- Time after shutdown
- Reactor vessel and containment status
- Vessel and core temperatures
- Fuel location
- Availability of required systems and support systems

This analysis addresses the BWRX-300 risk associated with a refueling outage. The systems modeled are evaluated based on anticipated activities associated with refueling operations.

To develop a suitable shutdown model, multiple bounding plant configurations/POSSs are defined with similar characteristics in relation to the residual heat and the availability of systems.

The outage plant operating mode and POS are used to define the initial plant condition for individual event sequence quantification.

Once the outage POSSs are defined, the duration of each is estimated to determine its contribution to the overall calculation of annual CDF. The duration is expressed in hours per refueling outage.

The Shutdown PSA considers outage plant configurations representative of the possible plant configurations during shutdown.

## **Initiating Events**

The IEs that challenge normal operation including the critical safety functions (e.g., heat removal, inventory control) during shutdown operations are determined. A shutdown IE is defined as any event that challenges normal operation and requires action to prevent core damage.

## **Event Trees**

The LPSD event tree construction considers the following aspects:

- Chronological order of system actuation
- Grouping of mitigating systems by safety functions

The event sequence analysis and end state nomenclature are the same as the at-power PSA. The associated success criteria analysis is performed similar to the at-power success criteria analysis. However, success criteria analysis for LPSD may be approximated by hand calculations.

## **System Analysis**

The unavailability of a system to perform its safety function on demand is evaluated by fault tree analysis similar to the fault tree analysis approach.

The necessary fault trees are identified following construction of the event trees. These fault trees represent the nodes included in the event trees and any required support system fault trees

Maximum use is made of the fault trees developed for the Full-Power PSA. Potential differences between the at-power and the shutdown fault tree models may result from differences in:

- Maintenance unavailabilities
- Success criteria between at-power and shutdown condition
- Initial system configuration between at-power and shutdown condition
- Human actions

Maintenance events used in the at-power model are not adjusted for the LPSD PSA, other than for POSs where some systems are entirely unavailable (for the POS). Additional maintenance events may be possible at LPSD that do not normally occur at full-power. The evaluation of maintenance for each POS is justified and documented.

## **Quantification**

The shutdown event sequence analysis models the effects on the following two critical safety functions during shutdown:

- Decay heat removal
- Containment during configuration changes in the containment boundary and reactor head during refueling

IE, associated frequencies, are identified that challenge the above critical safety functions. Event trees are developed specific to the shutdown configurations and the system fault tree analysis is based on at-power fault tree models adjusted to match shutdown conditions. The model development and quantification are performed using CAFTA and FTREX in a similar manner to the at-power PSA.

### **Start-up and Shutdown Transition Risks**

In the BWRX-300 PSA shutdown models supporting the design, the start-up and shutdown risks are included in the at-power conditions.

Planned shutdown and start-up transition states are difficult to model with PSA for several reasons. The reactor is under continuous transition, e.g., main circulation flow. Temperature and pressure in the reactor vary significantly during the operation. Several manual operator actions and decisions are required. In addition, the operation may require the use of many systems and/or components that are not used during power operation. Identification of risks is also more challenging as it requires detailed understanding of the systems related to process and control. For these reasons, certain assumptions and simplifications are performed to create a usable and practical PSA model.

The new event trees include the same IEs as the previous PSA model, but it also considers transients prior to control rod insertion and failure possibilities related to maneuvering of the control rods. Modeling and quantification of transition modes is similar to the at-power PSA, if quantified.

### **Key Inputs and Assumptions**

- At-Power BWRX-300 PSA models (note external events low power and shutdown models are developed in the external events sections)
- Representative BWRX-300 Refueling Outage Plan Key Assumptions
- Containment is open during the outage

#### **15.6.1.9 Spent Fuel Damage**

Spent fuel damage evaluation is required for the plant-specific BWRX-300 PSA because fuel damage frequency is an important contributor to release risk relative to other low risk contributors. For accidents, where the spent fuel is damaged outside the reactor core, the term spent fuel damage is applied. In particular, the fuel damage frequency may be an important contributor to release risk. The term “fuel damage” represents damage to the fuel outside the reactor vessel, while “core damage” is used for damage inside the vessel.

### **Methodology**

A separate PSA analysis investigates the event sequences leading to spent fuel damage. The analysis estimates the related event sequences and their frequencies, and it is documented separately as part of the PSA. The analysis covers both power operation and outages. Sabotage is explicitly excluded from the calculation of the spent fuel damage frequency.

During the performance of the BWR PSA, the Fuel Pool PSA is performed using the process and requirement of the LPSD PSA including the requirements in the LPSD PSA Standard. In this PSA, the fuel pool is analyzed as a separate POS, and the associated IEs are those that result in an event that challenges the normal operation of the fuel pool cooling or inventory requiring mitigation to prevent fuel damage. The overall process of event tree analysis, success criteria analysis, fault tree analysis, is the same process used in the LPSD PSA. Generally, the time available for responding to a fuel pool IE is much longer than the LPSD IE response.

Event sequences are developed and quantified that credit potential recovery actions taken by the operator.

### **Key Inputs**

- Site-specific external events

- Site-specific spent fuel handling equipment design
- Plant procedure for heavy load lifting and spent fuel handling
- Task outputs and preliminary results

This task develops the Fuel Damage PSA models. A number of IEs are expected to be screened with qualitative assessments.

#### **15.6.1.10 Systems Credited in the PSA**

The following design features are modeled in the PSA.

##### *Isolation Condenser System*

The ICS function modeled in the PSA is removing decay heat from the reactor by condensing steam in the ICS heat exchangers.

##### *Reactor Isolation Function*

The reactor isolation function mitigates the effects of large and medium sized pipe break LOCAs.

The term “RPV isolation system,” is used to refer to RIV closure. The system model is a surrogate for valves that are part of the NBS.

##### *Ultimate Pressure Regulation Function*

The ultimate pressure regulation function of the RPV provides emergency pressure relief in the event of a severe pressure transient.

##### *Control Rod Insertion Function*

The CRD system performs several insertion functions and consists of these design features: FMCRDs, HCU, and the CRD Hydraulic Subsystem.

The PSA-credited function of the control rod insertion is to insert negative reactivity into the reactor core rapidly upon a scram signal. Subcriticality is achieved with the negative reactivity insertion and terminates fission heat generation.

The CRD system also provides an RPV inventory makeup function.

##### *Feedwater Flow Reduction Function*

The feedwater flow reduction provides negative reactivity in a failure to scram condition by reducing feedwater flow resulting in increased core voiding (the BWRX-300 has a negative reactivity void coefficient).

##### *Containment Isolation Function*

The containment isolation function isolates containment in the event of accidents or other conditions and prevents the unfiltered radioactive release before they exceed allowable limits.

##### *Boron Injection System*

The BIS provides a separate, diverse means, D-in-D backup system to the CRD system for manually inserting negative reactivity into the reactor core for BDBAs. All equipment is located outside primary containment to allow easy access for testing and inspection activities during all plant operating conditions.

The BIS utilizes an aqueous solution of highly enriched sodium pentaborate decahydrate for reactivity control. The sodium pentaborate solution temperature is maintained above the solubility temperature due to the placement of the system within the reactor building.



### *CRD Injection Function*

The focus of this system for PSA is the CRD Hydraulic Subsystem that is used for inventory makeup and flow to the SDC pumps.

The CRD Hydraulic Subsystem provides clean, demineralized water that is regulated and distributed to provide charging to the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD Hydraulic Subsystem is also the source of purging water to the SDC system pumps and the NBS reactor water level reference leg instrument lines. Additionally, the CRD Hydraulic Subsystem provides high pressure injection to the reactor. This makeup water is supplied to the reactor via the drives.

### *Shutdown Cooling System*

The SDC system provides long-term decay heat removal after planned shutdowns. SDC consists of two independent trains designated as Train A and Train B. Each train suction is independently connected to an ICS condensate return line outside containment, downstream of the containment isolation valves.

### *Power Conversion Function*

There is only one function for the power conversion function modeled in the PSA. This function is providing water from the condenser, via the condensate pumps, feedwater heaters and feedwater pumps, to the reactor where it is heated, and steam is produced. Steam is transferred to the condenser via the turbine bypass valves.

Because the PSA model assumes a scram has occurred or is warranted, the full-power function of providing steam to the turbine for power generation is not considered.

### *Cooling Water Systems*

The function of PCW removes heat loads in the RB and TB.

PCW is a closed cooling water system supported by cooling from a portion of the Circulating Water System (CWS).

### *DC Power*

The electrical distribution system is an integrated power supply and distribution system for the power plant. Three plant systems constitute the overall electrical system: Safety Class 1 Electrical Distribution System, Safety Class 2 and 3 Electrical Distribution System and the Non-Safety Electrical Distribution System.

The electrical system powers automatic shutdown and decay heat removal functions. The Safety Class 1 portion of the electrical distribution system (includes the direct current (DC)) is limited to supplying power to Safety Class 1 SSCs within the RB rooms.

Safety Class 1 DC power three division (A, B, and C) arrangement supplies DC power to various loads. Each division has a DC battery and two redundant battery chargers powered from the Safety Class 2 and 3 Electrical Distribution System alternating current (AC) power system.

The primary load of the Safety Class 1 Electrical Distribution system is the Distributed Control and Information System (DCIS).

### *AC Power*

AC power modeled in the PSA, is providing medium and low voltage AC power to plant components.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The following seven AC buses are modeled:

1. Buses A1 and B1 provide 4160V AC power to BOP components such as feedwater pumps and condensate pumps. These buses are not backed by diesel generator.
2. Buses A2 and B2 provide 4160V AC power to other components, such as control rod drive pumps and reactor component cooling pumps. These buses are each backed by a standby diesel generator (SDG) in the event of LOPP.
3. Divisions 1, 2 and 3 provide 480V AC power to Motor Control Centers (MCCs), MOVs, smaller motors, and other plant equipment.

#### *Corium Shield*

The corium shield prevents core melt from damaging the containment liner once the core has broken through the bottom of the RPV.

The corium shield is a complementary design feature that includes a refractory material below the RPV. After core damage, in cases where the RPV is assumed to be at a low enough pressure to preclude direct containment heating, the core debris eventually migrates from the core region to the RPV lower head and exits the through a breach (e.g., lower head failure or CRD housing failure). The refractory material below the RPV prevents molten core concrete interaction and any potential ablation of the basemat and accompanying flammable gas generation.

#### *Passive Containment Cooling System*

The Passive Containment Cooling System (PCCS) rejects heat into the reactor cavity and/or equipment pool above containment. Supply and discharge connections from the pool are connected to closed loop piping within containment. Heat transfer occurs from the containment to the PCCS by natural convection and condensation.

#### *Containment Filtered Venting*

The containment pressure regulation function is venting pressure in the containment as a result of LOCAs, RPV pressure regulation, or core melt. By providing a vent to the containment, pressure, and temperatures are maintained. The release is filtered to reduce the amount of fission products.

Venting is required when no other containment heat removal method succeeds and in certain special situations where there is excessive containment pressure loads beyond the capacity of PCCS. Containment pressure control venting is required to prevent containment overpressurization due to non-condensable gas generation if passive containment heat removal fails. Venting requires operator action to open a pathway to the environment through a hardened vent.

#### *Heating Ventilation and Cooling System*

The HVAC system function is providing normal room atmospheric temperature and ventilation control and room cooling during accident scenarios.

Systems assumed to require HVAC for successful operation have placeholder transfer gates with an assumed failure point-estimate screening value used in PSA model quantifications.

### *Plant Pneumatic System*

The Plant Pneumatic System function provides motive and control air or nitrogen to various plant components. The pneumatic system supports multiple systems valves.

#### **15.6.1.11 Level 2 PSA**

The Level 2 PSA confirms that the BWRX-300 plant for Darlington meets the SRF and LRF safety goals specified by the CNSC.

The development steps of the Level 2 PSA (NEDC-33946P Reference 15.6-25) are:

- Develop an interface between the Level 1 and Level 2 PSA
- Identify and model safety functions and operator actions
- Perform the containment performance analysis
- Perform the accident progression analysis and develop the Containment Event Trees (CETs)
- Perform the source term analysis
- Quantify the model and interpret results

#### **Interface Between Level 1 and Level 2 PSA Event Tree Definition**

The first step in developing the Level 2 PSA is assigning damage classes to the accident sequences resulting in core damage. The BWRX-300 PSA Level 1 Event Tree analysis provides definitions for five damage classes in addition to the successful mitigation state. Table 15.6-6 provides the Level 1 Probabilistic Safety Assessment Damage Class Definitions used in this analysis.

While not strictly required for assessing small and large release frequencies, some element of release timing is also considered in the analysis. For example, if ICS fails to operate during transients, the core can remain adequately cooled for a time by injecting water through the CRD system. If the CRD water inventory is insufficient over the long-term, and if SDC is not initiated before the CRD suction source is depleted, the core can no longer be cooled, and core damage may occur. This is expected to occur significantly later than if CRD did not succeed at all. For this reason, a 'late' designator is appended to damage classes where core damage is significantly delayed. It is expected that this early/late distinction could facilitate applications concerned with dose consequence analysis of reactor accidents.

### *Core Damage Sequences*

Aspects of core damage sequences that are relevant to the potential for radionuclide release characterization are carried forward to the Level 2 PSA. These include:

- RPV pressure at the time of core damage
- Timing of core damage
- Availability/failure of mitigating measures
- Timing of containment failure (relative to core damage)
- Availability of containment isolation

### *RPV Pressure at Core Damage*

If RPV pressure is not lowered substantially during the accident progression, then the potential exists for pressure-driven debris ejection at the time of vessel breach. The resultant fragmentation of the debris may increase the heat transfer to the containment quickly, resulting in a rapid containment heat-up, and pressurization. This direct containment heating is expected to result in containment failure. The potential for this phenomenon warrants development of a CET. The core damage state for high and low RPV pressure are delineated by the accident class. This interface is addressed by developing separate CETs that account for RPV pressure.

### *Timing of Core Damage*

If the core damage occurs very late in the sequence, then these scenarios are given a 'late' designation. This interface is addressed by developing separate CETs that assume core damage occurs very late. The significance of late core damage relative to early core damage scenarios is that protective measures are taken to reduce the impact (e.g., dose to the public) of SA. No quantitative surrogate is developed for what constitutes early versus late damage. Qualitatively, late core damage is assigned for those sequences where safety functions are met initially but fail after a moderate amount of elapsed time (judged to be several hours).

### *Availability/Failure of Mitigating Measures*

Mitigating measures (i.e., equipment and operator actions) and their failures are addressed via fault trees. This interface is addressed inherently by coupling the Level 1 sequences with the Level 2 model so that any basic events relevant to both core damage and radionuclide release will propagate to both levels of the PSA model. To the extent that is possible, the same dependency modeling and basic events are used in both models.

### *Timing of Containment Failure*

If containment is breached or severely degraded as part of the phenomenology of sequence leading to core damage, this interface is addressed by assigning a core damage class that reflects this feature of the sequence.

### *Availability of Containment Isolation*

If core damage occurs as part of a failure to isolate the containment, this interface is addressed by developing a dedicated CET specifically for that class of core damage.

### *Level 2 Mitigation*

The systems and human actions used in the PSA model were taken from the Level 1 PSA (that contains aspects of containment performance, such as containment venting) and no new system or operator actions are developed. The relevant functions to the Level 2 PSA analysis are:

- Late inventory injection
- Containment isolation
- Corium shield
- PCCS
- Containment venting
- RPV depressurization

The effects these functions have on the Level 2 accident sequences is described in the Level 2 Accident Sequence Analysis because their impact on radionuclide release and retention is sequence- specific.

### *Containment Performance*

The following containment failure modes are considered in the Level 2 PSA:

- Overpressure
- Failure of containment isolation
- Basemat melt-through
- Hydrogen deflagration
- Venting
- Steam explosion
- Containment bypass
- Direct containment heating

### *Containment Overpressure*

No detailed ultimate containment performance assessment is available at this stage in the BWRX-300 design. However, the containment failure pressure is assumed to be 1.5 MPa (210 psig) for this evaluation. If the containment vent fails to operate, then the containment is assumed to fail.

Fail-to-scram sequences result in severe dynamic forces inside containment during RPV failure that result in gross containment failure.

### *Containment Isolation/Bypass*

Each core damage scenario requires containment isolation. For core damage scenarios resulting from breaks outside containment (e.g., breaks in the main steam lines outside containment), that the core damage event results in a simultaneous large release due to the inability to isolate containment.

### *Basemat Melt-Through*

BWRX-300 is equipped with a corium shield that covers the basemat. This corium shield prevents MCCI. However, for the corium shield to function, the downward heat flux between the ex-vessel debris bed must be limited by overlying water pool. If debris is not covered with water, then the limited upward heat flux and steaming results in failure to quench the molten debris pool and results in a containment basemat melt-through.

### *Hydrogen Deflagration*

During the SA progression, cladding oxidation results in significant hydrogen generation. If the accident occurs when containment is not inerted (e.g., purging nitrogen from containment prior to a refueling outage), then it is assumed that the hydrogen will ignite, and the dynamic forces result in containment failure.

### *Containment Venting*

In scenarios where the containment pressure is increases beyond the design pressure, the containment vent is initiated. The vent is equipped with a filter. Failure to vent when required is results in containment overpressure.

### *Direct Containment Heating*

Direct containment heating occurs in a SA if the RPV fails while at high pressure. The resultant failure is accompanied by fragmentation of debris and entrainment in any effluent post-RPV

rupture. The increased surface area of the debris results in rapid heat-up and pressurization of containment that leads to gross containment failure.

### *Steam Explosion*

If an unquenched molten pool of corium comes into contact with a pool of water, the potential exists for there to be a rapid boiling event, further fragmenting the corium, increasing the heat transfer surface between debris and water. The fragmentation and entrainment of corium and rapid boiling and expansion of steam and water is a steam explosion. The dynamic effects of this event result in gross containment failure. The likelihood of this event to occur due to reflooding in-vessel is judged to be very small. The likelihood of the event to occur due to vessel failure and debris entry into an underlying pool of water is small, but more likely than in-vessel. While the dynamics of lower head failure are uncertain, it is likely that low pressure vessel failure results in fractional corium relocation to the underlying pool rather than a total “dump” of the entire corium pool. By limiting the amount of corium that relocates to the lower pool at one time, the potential for steam explosions is reduced.

This section describes the Level 2 accident sequences for each CET.

### **Class I/I-L**

Core damage sequences that occur with the RPV at low pressure and that nominally have containment intact are Class I. There are core damage sequences that occur in Class I core damage sequences that occur “late”. As a result a CET is developed for both “early” and “late” Class I core damage. The node descriptions in this section apply to the “early” Class I core damage scenarios. Late core damage sequences have the same general progression, but any early release categories are changed to late releases for late core damage events.

### **Class II**

The containment isolation function is reassigned to the Level 2 CETs and given the definition of Class II core damage (core damage resulting from containment failure due to overpressure). In the BWR operating fleet, Class II core damage may result from injection without containment decay heat removal. The resultant overpressure of containment leads to failure and loss of the injection source due to loss of net positive suction head (NPSH), adverse equipment environments, and disruption of injection lines. Core cooling is provided by ICS, and no separate containment heat removal method is required. If core cooling is provided by CRD, then there must be a letdown path outside containment established to remove inventory heated in the RPV. So, failure of decay heat removal results in core damage due to core cooling failures before containment failure for evaluated sequences in the BWRX-300 PSA.

### **Class III/III-L**

Core damage sequences that occur with the RPV at high pressure and that nominally have containment intact are Class III. There are core damage sequences that occur with Class III that occur “late”, and as a result, CET is developed for both “early” and “late” Class III core damage. The node descriptions in this section apply to the “early” Class III core damage scenarios. Late core damage sequences have the same general progression, but early release categories are changed to late releases for late core damage events.

### **Class IV**

Class IV core damage scenarios are failure to scram scenarios. If core damage occurs from failure to scram scenarios, SA progression results in more severe RPV and containment conditions than those scenarios with successful negative reactivity insertion. Generally, a core damage scenario for failure to scram in a BWRX-300 is expected to have core damage occurring

at elevated RPV pressure or pre-core damage energetic RPV failure due to overpressure/overpower. The containment loading resulting from these scenarios is expected to exceed the containment capacity and gross failure of containment is expected to occur.

### **Class V**

Class V sequences are those with an initial bypass of containment. This containment bypass provides a radionuclide release path, so any Class V sequences result in large early releases.

### **Class VR**

The VR core damage sequences involve excessive LOCAs in the mid and lower vessel regions and successful operation of the containment vent, but due to the size and location of the breaks core damage cannot be prevented.

#### **15.6.1.11.1 Source Term Analysis**

Two releases are of special interest in the source term analysis: small and large releases. Small and large releases are defined in CNSC REGDOC-2.5.2 (Reference 15.6-9). Small releases are considered those that release more than  $10^{15}$  Bq of Iodine-131 to the environment. Large releases are those that release more than  $10^{14}$  Bq of Cesium-137 to the environment. Sequences that meet the criteria for both small and large releases are counted as large releases, rather than both. Source terms are determined using MAAP. Release calculations are performed in MAAP for representative sequences. Release categories are then assigned to those without specific MAAP runs based on the calculations performed. This approach ensures that significant scenarios with realistic release categories are described in detail in the Level 2 PSA.

MAAP generally plots radionuclide release as a fraction of total inventory for select "groups" of fission products. The groups of fission products are subdivided by molecular compound and individual isotopic transport is not specifically tracked. However, MAAP provides the requisite information to understand the fractional distribution of all isotopes of a given element across fission product groups. It is assumed that all isotopes of a given element behave identically for the purposes of group assignment and subsequent transport. For example, if 50% of element X is in Fission Product Group 1, and all of group 1 is released, then 50% of every isotope of X is released in group 1.

Refer to Chapter 3, Appendix 3E for additional discussion regarding the MAAP computer code.

#### **15.6.1.12 Model Integration and Quantification**

The Level 2 PSA model is integrated with the Level 1 PSA model. The Level 1 accident sequences are placed under "collector gates" that group core damage sequences into classes.

The Level 2 ETs are constructed with the entry branch name identical to the applicable Level 1 Core Damage class collector gate. Functional node branches are given names identical to those in the functional fault tree file. When the event trees are converted into fault tree logic (creating accident sequence logic) and merged with functional fault tree logic, an integrated, quantifiable fault tree is developed.

The fault tree is quantified using the following PSA model development codes (see Chapter 3, Appendix 3E):

- CAFTA
- PRA Quant

In addition, FTREX 1.8 is used as the quantification/cutset generation engine and SysImp is used to develop importance measures for systems, components, and HFEs.

Intact, small release, and large release frequencies are quantified at a truncation of 1E-15/yr. The cutsets are examined to ensure there was <5% top event frequency drop if the truncation limit are set to 1E-14/yr. This provides evidence that no risk-significant sequences were truncated.

## **15.6.2 Results of the Level 1 Probabilistic Safety Assessment**

### **15.6.2.1 Initiating Events**

The IEs were derived for internal events at-power and for low power shutdown states. The IEs are tabulated with descriptions and frequencies in Table 15.6-7 and Table 15.6-8.

### **15.6.2.2 Mitigating Systems Modeled**

The following front-line mitigating systems are modeled to mitigate the IEs. These mitigating systems or functions are:

- Shutdown Cooling (SDC)
- Isolation Condenser System (ICS)
- Containment Isolation (CI)
- Ultimate Pressure Relief
- Control Rod Insertion
- Boron Injection
- Control Rod Injection makeup
- Corium Shield
- Containment Venting
- Passive Containment Cooling System (PCCS)
- Cooling Water System
- DC Power
- Feedwater Runback
- Reactor Pressure Vessel (RPV) Isolation
- AC Power
- Heating Ventilation and Cooling
- Plant Pneumatics

### **Level 1 Event Tree Endpoints**

The Level 1 PSA event tree end points end in one success or five core damage states. The five core damage states are described below.

Event sequences are qualitatively analyzed to determine which sequences lead to core damage end states. The core damage sequences are grouped together based upon the overall challenge to the containment barrier and defined as:

1. Containment state: intact or breached
2. RPV pressure



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

3. Energy deposition into containment
4. Containment bypass

The above considerations result in the following event tree end points:

1. OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
2. CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.
3. CD II: The containment is breached, either due to overpressurization or venting, while the core is successfully cooled. Core damage results from failure of maintaining long-term heat removal.
4. CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.
5. CD IV: CD IV: Core damage results from an accident sequence with an initial failure of effective reactivity control (failure to scram due to failure of RPS, control rod binding, etc.). This has the potential to affect containment more severely than the CD I and CD III because more energy is deposited into containment prior to RPV failure. All CD IV end states could be treated as CD I or CD III (depending on the RPV pressure) without affecting the results of the containment analysis. This end state is retained in this scoping analysis to more readily allow for sensitivity analyses related to reactivity control.
6. CD V: The containment is bypassed at the time of core damage.
7. CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

**Event Tree Transition – Level 1 to Level 2 PSA**

The core damage Level 2 PSA accident sequence analysis is:

1. Class 1/I-L  
Core damage sequences occur with the RPV at low pressure and nominally have containment intact are Class I. There are core damage sequences that occur with Class I that occur “late” and CET is developed for both “early” and “late” Class I core damage.
2. Class II  
Class II core damage (core damage resulting from containment failure due to overpressure). Class II core damage often results from injection without containment decay heat removal. The resultant overpressure of the containment leads to failure and loss of the injection due to loss of NPSH, adverse equipment environments, and disruption of injection lines similarly to the Level 1 PSA Core Damage Frequencies.
3. Class III/III-L  
Core damage sequences that occur with the RPV at high pressure and that nominally have containment intact are Class III. There are core damage sequences that occur with Class III that occur “late”, and a CET is developed for both “early” and “late” Class III core damage.

4. Class IV

Class IV core damage scenarios are failure to scram and failure to depressurize scenarios. If this core damage scenario occurs, SA progression results in more severe RPV and containment conditions than those scenarios with successful negative reactivity insertion. The containment loading resulting from these scenarios is expected to exceed the containment capacity and gross failure of the containment is expected to occur.

5. Class V

Class V sequences are those with core damage and an initial containment bypass. Containment bypass provides a radionuclide release path. Any Class V sequences result in large early releases.

6. Class VR

Core damage occurs due to RPV ruptures in the lower or mid-vessel regions, the containment vent functions.

### **Small Release Frequencies**

The SRFs for the Level 2 PSA are not reported, because achieving a small release without achieving large release is within a very tight release band. If a sequence becomes SRF, it is expected the SRF sequence deteriorates to LRF.

### **Large Release Frequencies**

The LRF for the Level 2 PSA At-Power, Shutdown and Low Power, seismic, fire, high wind and internal flood are listed Table 15.7.9-2.

### **15.6.3 Probabilistic Safety Assessment Insights and Applications**

The Darlington BWRX-300 PSA activities are ongoing as the design progresses from conceptual to detailed. The PSA scope includes:

- Level 1 PSA
  - Internal Events at-Power and Low Power Shutdown states
  - External Events at-Power and Low Power Shutdown states, (seismic fire, high wind, and flood)
- Level 2 PSA At-Power and Low Power and Shutdown

The CNSC REGDOC-2.4.2, Section 3 (Reference 15.6-1) objectives (a thru h) are:

- a. Provide a systematic analysis
- b. Demonstrate a balanced design that no particular feature or postulated IE makes a disproportionate contribution to overall risk
- c. Provide confidence of no cliff-edge effects
- d. Provide assessments of quantitative safety goals
- e. Provide site-specific assessments of external hazards
- f. Identify facility vulnerabilities and systems where design improvements or modifications to operational procedures could reduce probabilities of severe accidents or mitigate consequences

- g. Assess adequacy of emergency operating procedures
- h. Provide insights into severe accident management

The BWRX-300 PSA assesses the design to meet the CNSC REGDOC-2.4.2 (Reference 15.6-1) objectives. PSA objectives (a to f) are assessed at this stage of the design and objectives (g and h) are assessed once the detailed design and operational procedures are available.

Concern that small changes can lead to large differences in plant effects – i.e., cliff-edge effects are largely ameliorated by the demonstrated margin between plant risk metrics and the CNSC REGDOC-2.5.2 (Reference 15.6-9) safety goals. Within the PSA accident sequence and success criteria analysis, no cliff-edge effects were discovered where plant parameters approach critical thresholds where accident sequences are expected to follow a significantly more severe pathway. However, uncertainty in these analyses will be formally explored in more detail in future PSA work.

Tables 15.7.9-1 and 15.7.9-2 show that the PSA CDF and LRF results to date meet the CNSC safety goals.

CNSC REGDOC-1.1.2, Section 4.4.4 (Reference 15.6-8) states that hazards analyses be performed. As presented in this section, internal and external hazards are identified and screened for PSA. Natural external hazards and human-induced external and internal hazards are assessed. There is limited assessment of external and internal hazards interaction. A more detailed hazard combination assessment is performed commensurate with the design progress.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) states that a PSA be included in the Licence to Construct application. The PSA identifies facility vulnerabilities. The PSA application should also:

1. Verify EOPs are adequate during commissioning and future operation
2. Provide insights into the Severe Accident Management (SAM) program
3. Describe how PSA is used during commissioning and future operation

Based on the PSA work to date, CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) requirements are not met at this time because Operating Manuals, EOPs and SAM program are not defined. The PSA is updated when the design matures and operational documents including Operating Manuals, EOPs, and SAM program are developed.

PSA is further updated for the Operating Licence Application as the design is finalized and operational information matures.

#### **15.6.3.1 Summary**

The PSA CDF, SRF, and LRF results to date meet the CNSC safety goals.

Section 15.6 is a high-level summary of the Internal and External Event Hazard Screening, and Level 1 and Level 2 PSA status supporting the BWRX-300 PSAR.

The PSA scope:

1. Level 1 PSA Internal At-Power, Low Power Shutdown States
2. Level 1 PSA for external events includes a power seismic and high wind analysis
3. Level 2 PSA At-Power

The CNSC REGDOC-2.4.2, Section 3 (Reference 15.6-1) objectives are reviewed and six of the eight objectives of CNSC REGDOC-2.4.2 (Reference 15.6-1) are met. Two other objectives (adequacy of EOPs and insights on SAM) are not assessed because the plant design is conceptual and the EOP procedures and SAM program are prepared as the design is finalized.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

CNSC REGDOC-1.1.2, Section 4.4.4 (Reference 15.6-8) requires a hazard analysis. Internal and external hazards are identified and screened. Natural external hazards, human-induced external and internal hazards are assessed.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) states a PSA is conducted as required by CNSC REGDOC-2.4.2 (Reference 15.6-1) and includes:

1. Verification that EOPs are adequate during commissioning and future activities
2. Provide insights into severe accident management program
3. Describe how PSA is used during commissioning and future operation to identify design improvements, modifications to operational procedures to reduce probabilities of severe accidents and mitigate consequences.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) requirements are addressed after Operating Manuals, EOPs and SAM programs are defined, and the PSA is updated.

The PSA CDF, SRF, and LRF results “to date” indicate that the CNSC safety goals are met.

The SRF is not reported, because achieving a small release without achieving large release would be within a very tight release band. If a sequence becomes SRF, it is expected the SRF sequence deteriorates to an LRF. The LRF requirement is more restrictive than SRF. Therefore, SRF is conservatively grouped with LRF.

The CDF, Small Release Frequency, and LRF results meet the CNSC safety goals as shown in Tables 15.7.9-1 and 15.7.9-2.

#### **15.6.4 References**

- 15.6-1 CNSC Regulatory Document REGDOC-2.4.2, “Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.”
- 15.6-2 ANSI/ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment For Advanced Non-Light Water Reactor Nuclear Power Plants Publication,” American Nuclear Society.
- 15.6-3 USNRC Regulatory Guide 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities.”
- 15.6-4 USNRC Regulatory Guide 1.206, “Applications for New Power Plants.”
- 15.6-5 IAEA Safety Standards Series No. SSG-3, “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants,” International Atomic Energy Agency.
- 15.6-6 IAEA Safety Standards No. SSG-4, “Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Plants,” International Atomic Energy Agency.
- 15.6-7 CSA N290.17-17, “Probabilistic safety assessment for nuclear power plants,” CSA Group.
- 15.6-8 CNSC Regulatory Document REGDOC-1.1.2, “Licence Application Guide: Licence to Construct a Reactor Facility.”
- 15.6-9 CNSC Regulatory Document REGDOC-2.5.2, “Design of Reactor Facilities: Nuclear Power Plants.”
- 15.6-10 IAEA-TECDOC-1106, “Living Probabilistic Safety Assessment (LSPA),” International Atomic Energy Agency.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

- 15.6-11 IAEA Safety Standards Series No. SSG-64, "Protection against Internal Hazards in the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 15.6-12 ANSI/ASME/ANS RA-S-1.1-2022, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment For Nuclear Power Plants," American Nuclear Society.
- 15.6-13 IAEA-TECDOC-1804, "Attributes of Full Scope Level 1 Probabilistic Safety Assessment for Applications in Nuclear Power Plants," International Atomic Energy Agency.
- 15.6-14 USNRC NUREG/CR-6451, BNL-52498, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants."
- 15.6-15 NEA/CSNI/R (2009)4, "Probabilistic Safety Analysis (PSA) of Other External Events Other Than Earthquake," Nuclear Energy Agency.
- 15.6-16 USNRC Backgrounder, "Drones and Nuclear Power Plant Security," ADAMS Accession Number ML20300A547.
- 15.6-17 IAEA-TECDOC-719, "Purpose of Probabilistic Safety Assessment," International Atomic Energy Agency.
- 15.6-18 USNRC Web Page "<https://nrc.nrel.gov/publicdocs/CCF/ccfparamest2015.pdf>."
- 15.6-19 USNRC NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants."
- 15.6-20 USNRC NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities."
- 15.6-21 IAEA-TECDOC-1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 15.6-22 USNRC NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)."
- 15.6-23 EPRI 3002000709, "Seismic Probabilistic Risk Assessment Implementation Guide," Electric Power Research Institute.
- 15.6-24 IAEA-TECDOC-1144, "Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes," International Atomic Energy Agency.
- 15.6-25 NEDC-33946P,"BWRX-300 Darlington New Nuclear Project (DNNP) Probabilistic Safety Assessment Methodology," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.6-26 NK38-REP-03611-10043 R003, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 15.6-27 USNRC NUREG/CR-6850 Supplement 1 (EPRI 1019259), "Fire Probabilistic Risk Assessment Methods Enhancements."
- 15.6-28 USNRC NUREG-2169 (EPRI 3002002936), "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database."
- 15.6-29 USNRC NUREG/CR-7150 (EPRI 1026424), "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure."

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

15.6-30 USNRC NUREG/CR-7150 (EPRI 3002001989), "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE), Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Fire."

**Table 15.6-1: Internal Hazard Qualitative Screening Analysis**

<b>Hazard</b>	<b>Screening Criterion (Refer to Subsection 15.6.1.2)</b>
Heavy Load Drop	Not screened qualitatively
Release of Chemicals from On-Site Storage	1
Turbine-Generated Missiles	Not screened qualitatively
Other Internally-Generated Missiles	1, 2
Fires	Not screened qualitatively. Covered by BWRX-300 Internal Fire Scoping Evaluation
Explosions	Not screened qualitatively. Covered by BWRX-300 Internal Fire Scoping Evaluation
Collapse of Structures	4, 5
Pipe Whips	4
Jet Effects	4
Internal Flooding	Not subject to screening

**Table 15.6-2: Internal Hazard Quantitative Screening Analysis**

<b>Hazard</b>	<b>Quantitative Screening Results</b>
Turbine-Generated Missiles	Screened from further PSA analysis
Internal Flooding	Covered by Internal Flooding PSA
Heavy Load Drop	Not screened out quantitatively



**Table 15.6-3: External Hazards Qualitatively Screened Out of the PSA**

Hazard Type	Screen Qualitatively
Individual Hazards	Extreme Snow, Extreme Hail, Mist, White Frost, Drought, Salt Storm, Sand Storm, Lightning, Explosion Outside Plant, Explosion after Pipeline Accident, Chemical Release Outside or Inside Site, Chemical Release after Transportation Accident, Chemical Release after Pipeline Accident, Magnetic Disturbance, Land Rise, Soil Frost, Animals, Volcanic Phenomena, Avalanche, Above-Water Landslide, External Fire, Excavation Work, Direct Impact from heavy transportation within Site, Missiles from military activity, Internal Fire Spreading from Other Plant, Contamination from Chemicals, Strong Water Current, Low Water Level, High Water Temperature, Low Water Temperature, Underwater Landslide, Surface Ice, Frazil Ice, Organic Material in Water, Corrosion from Salt Water, Solid or Fluid Impurities from Ship Release, Chemical Release to Water, Direct Impact from Ship Collision.
Co-Existent Hazards	High Air and Water Temperature; Low Air and Low Water Temperature; Extreme Rain and Lightning; Drought and Low Water Level; Drought and External Fire; Explosion Outside Plant and Chemical Release; Explosion after Transportation Accident and Chemical Release after Transportation Accident; Explosion after Pipeline Accident and Chemical Release after Pipeline Accident; Chemical Release and Contamination from Chemicals; Extreme Snow and Earthquake.

**Table 15.6-4: External Hazards Quantitative Assessment Summary**

Hazard	Initiating Event Frequency (/yr)	CCDP	CDF / LRF (/yr)	Quantitative Screening
Seismic	Ground motion bin frequencies are assigned by the seismic PSA based on the seismic hazard	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
Internal Fire	Fire scenario frequencies are assigned by the fire PSA	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
High Wind, including high wind hazard combinations: Strong Wind and Extreme Air Pressure; Strong Wind and Ice barriers; Tornadoes and Extreme hail; Wind-Driven Precipitation	Wind speed bin frequencies are assigned by the high wind PSA based on the straight wind and tornado hazards	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
High Air Temperature	5.0E-2	3.1E-9	1.5E-10	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Low Air Temperature	2.8E-3	3.1E-9	8.5E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Meteorite	5.3E-10	1.0E+0	5.3E-10	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Explosion after Transportation Accident	3.0E-6	4.1E-7	1.2E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Hazard	Initiating Event Frequency (/yr)	CCDP	CDF / LRF (/yr)	Quantitative Screening
Satellite Crash	7.8E-8	4.1E-7	3.2E-14	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Airplane crash	3.4E-12	1.0E+0	3.4E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
High Water Level	1.0E-2	3.1E-9	3.1E-11	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.

(1) The total contribution for quantitatively screened external hazards is also less than 5% of the total 1E-7/yr CDF and LRF assumed for the BWRX-300.

**Table 15.6-5: Generic List of Boiling Water Reactor Transient Initiating Events**

No.	Initiating Event
1	Electric load rejection
2	Electric load rejection with TBV failure
3	Turbine trip
4	Turbine trip with TBV failure
5	MSIV closure
6	Inadvertent closure of one MSIV
7	Partial MSIV closure
8	Loss of normal condenser vacuum
9	Pressure regulator fails open
10	Pressure regulator fails closed
11	Inadvertent opening of a safety/relief valve (stuck)
12	Turbine bypass fails open
13	Turbine bypass or control valves cause increased pressure (closed)
14	Recirculation control failure - increasing flow
15	Recirculation control failure - decreasing flow
16	Trip of one recirculation pump
17	Trip of all recirculation pumps
18	Abnormal start-up of idle recirculation pump
19	Recirculation pump seizure
20	Feedwater - increasing flow at-power
21	Loss of feed water heater
22	Loss of all feed water flow
23	Trip of one feed water pump (or condensate pump)
24	Feedwater - low flow
25	Low feedwater flow during start-up or shutdown
26	High feedwater flow during start-up or shutdown
27	Rod withdrawal at-power
28	High flux due to rod withdrawal at start-up
29	Inadvertent insertion of rod or rods
30	Detected fault in reactor protection system
31	Loss of offsite power
32	Loss of auxiliary power (loss of auxiliary transformer)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

No.	Initiating Event
33	Inadvertent start-up of Inadvertent start-up of High Pressure Core Injection System/ High Pressure Core Spray System
34	Scram due to plant occurrences
35	Spurious trip via instrumentation, RPS fault
36	Manual scram- no out-of-tolerance condition
37	Cause unknown

**Table 15.6-6: Level 1 PSA Damage Class Definitions**

<b>Class</b>	<b>Definition</b>
I	The containment is nominally intact when core damage occurs. Core damage occurs with the RPV at low/controlled pressure.
II	The containment is breached due to overpressurization or venting during the accident, but before core damage occurs. Core damage occurs due to failure of long-term decay heat removal post-containment failure.
III	The containment is nominally intact when core damage occurs. Core damage occurs with the RPV at high pressure.
IV	Failure to scram occurs with failure to insert sufficient negative reactivity prior to core damage.
V	Core damage occurs with the containment bypassed.
VR	Excessive LOCAs in mid and lower vessel regions and successful operation of containment vent but due to size and location of the breaks, core damage cannot be prevented.

**Table 15.6-7: Internal Initiating Events At-Power**

IE Name Label	Description	Frequency (per RX yr.)	Comments
%TRANS	General Transient	7.4E-01	
%COND	Loss of Condenser Heat Sink	1.1E-01	Includes MSCIV and MSRIV Closure and Loss of Circulating Water
%FW	Loss of Feedwater (ALL)	6.0E-02	
%LOPP	Loss of offsite power, all categories, power operation	3.1E-02	This is the total frequency of all LOPP events and represents the other "LOOP-IEs" listed here
%LOPP-G	Loss of offsite power, grid-related, power operation	1.1E-02	
%LOPP-P	Loss of offsite power, plant-centered, power operation	2.0E-03	
%LOPP-S	Loss of offsite power, switchyard-centered, power operation	1.3E-02	
%LOPP-W	Loss of offsite power, weather-related, power operation	6.0E-03	
%MSLB-OUT	Main Steam Line BOC	3.0 E-03	MS Break Inside containment contained within the LOCA Frequency
%FLB-Out	Feedwater Line BOC	6.0E-04	FW Break Inside containment contained within the LOCA Frequency
%E-LOCA	Excessive LOCA	7.0E-09	Includes Vessel Rupture
%LLOCA-I	Large LOCA – Isolable	3.6E-06	
%LLOCA-N	Large LOCA – Non-isolable	6.1E-07	
%MLOCA-I	Medium LOCA – Isolable	3.1E-05	
%MLOCA-N	Medium LOCA – Non-isolable	1.6E-06	
%SLOCA-S-I	Small LOCA Steam – Isolable	1.6E-05	
%SLOCA-S-N	Small LOCA Steam – Non-isolable	2.6E-07	
%VSLOCA	Very-Small LOCA	3.4E-03	

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

IE Name Label	Description	Frequency (per RX yr.)	Comments
%SLOCA-W-I	Small LOCA Water– Isolable	6.7E-05	
%SLOCA-W-N	Small LOCA Water – non-isolable	2.5E-07	
%A1	Loss of 4160V AC Bus A1	3.4E-03	Reactor Trip
%B1	Loss of 4160V AC Bus B1	3.4E-03	Reactor Trip
%A2	Loss of 4160V AC Bus A2	3.4E-03	Manual Shutdown
%B2	Loss of 4160V AC Bus B2	3.4E-03	Manual Shutdown
%A2-U	Loss of 4160V AC Bus A2-Unknown	3.4E-03	Manual Shutdown
%B2-U	Loss of 4160V AC Bus B2-Unknown	3.4E-03	Manual Shutdown
%DIV1	Loss of 480V AC Bus Division 1 (4 buses total)	2.3E-02	Manual Shutdown
%EB-LGA	Loss of Bus EB LGA	5.7E-03	Manual Shutdown
%DIV2	Loss of 480V AC Bus Division 2 (4 buses total)	2.0E-02	Manual Shutdown
%EB-LGB	Loss of Bus EB LGB	6.0E-03	Manual Shutdown
%DIV3	Loss of 480V AC Bus Division 3 (4 buses total)	2.2E-02	Manual Shutdown
%VDC-A	Loss of Bus 250V DC-A	1.0E-03	Manual Shutdown
%VDC-B	Loss of Bus 250V DC-B	1.0E-03	Manual Shutdown
%VDC-C	Loss of Bus 250V DC-C	1.0E-03	
%PWC-All	Loss of All PCW	2.0E-04	
%PCW-A	Loss of A Train PCW	1.8E-03	
%PCW-B	Loss of B Train PCW	1.8E-03	
%IA	Loss of All Instrument Air	7.2E-03	



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.6-8: Internal Initiating Events At-Power and Shutdown**

POS -->		At-Power Per Calendar Year	Hot Shutdown and Stable Shutdown	Cold Shutdown	Refueling (reactor cavity drained)	Refueling (reactor flooded to normal with the fuel pool gate installed)	Refueling (reactor flooded to normal with fuel pool gate removed)
Time in POS%		90	0.11	0.11	0.4	0.3	4.4
Initiator	Description	Mean	Mean	Mean	Mean	Mean	Mean
%LOOP-G	LOOP, grid-related, power operation	9.90E-03	1.10E-05	1.21E-04	4.40E-05	3.30E-05	4.84E-04
%LOOP-P	LOOP, plant-centered, power operation	1.80E-03	2.00E-06	2.20E-05	8.00E-06	6.00E-06	8.80E-05
%LOOP-S	LOOP, switchyard-centered, power operation	1.20E-02	1.33E-05	1.47E-04	5.33E-05	4.00E-05	5.87E-04
%LOOP-W	LOOP, weather-related, power operation	5.40E-03	6.00E-06	6.60E-05	2.40E-05	1.80E-05	2.64E-04
%MSLB-OUT	Main Steam Line Break Outside Containment	2.70E-03	3.00E-06	N/A	N/A	N/A	N/A
%FLB-OUT	FW Line Break Outside Containment	5.40E-04	6.00E-07	N/A	N/A	N/A	N/A
%E-LOCA	Excessive LOCA	7.0E-09	7.8E-12	8.6E-11	3.1E-11	2.3E-11	3.4E-10
%LLOCA-I	Large LOCA – Isolable	3.3E-06	3.7E-09	N/A	N/A	N/A	N/A
%LLOCA-N	Large LOCA – Non-isolable	5.5E-07	6.1E-10	N/A	N/A	N/A	N/A
%MLOCA-I	Medium LOCA – Isolable	2.8E-05	3.1E-08	N/A	N/A	N/A	N/A
%MLOCA-N	Medium LOCA – Non-isolable	1.4E-05	1.6E-08	N/A	N/A	N/A	N/A
%SLOCA-S-I	Small LOCA Steam – Isolable	2.3E-05	2.6E-08	N/A	N/A	N/A	N/A
%SLOCA-S-N	Small LOCA Steam – Non-isolable	1.4E-07	1.6E-10	N/A	N/A	N/A	N/A
%VSLOCA	Very-Small LOCA	3.1E-03	3.4E-06	N/A	N/A	N/A	N/A

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

POS -->		At-Power Per Calendar Year	Hot Shutdown and Stable Shutdown	Cold Shutdown	Refueling (reactor cavity drained)	Refueling (reactor flooded to normal with the fuel pool gate installed)	Refueling (reactor flooded to normal with fuel pool gate removed)
Time in POS%		90	0.11	0.11	0.4	0.3	4.4
Initiator	Description	Mean	Mean	Mean	Mean	Mean	Mean
%SLOCA-W-I	Small LOCA Water– Isolable	6.0E-05	6.7E-08	N/A	N/A	N/A	N/A
%SLOCA-W-N	Small LOCA Water – non- isolable	2.3E-07	2.6E-10	N/A	N/A	N/A	N/A
%A2	Loss of 4160V AC Bus A2	5.2E-03	5.8E-06	6.4E-05	2.3E-05	1.7E-05	2.5E-04
%B2	Loss of 4160V AC Bus B2	5.2E-03	5.8E-06	6.4E-05	2.31E-05	1.7E-05	2.5E-04
%DIV1	Loss of 480V AC Bus Division 1 (four buses total)	3.1E-03	3.4E-06	3.8E-05	1.4E-05	1.0E-05	1.5E-04
%DIV2	Loss of 480V AC Bus Division 2 (four buses total)	3.1E-03	3.4E-06	3.8E-05	1.4E-05	1.0E-05	1.5E-04
%PCW-All	Loss of All PCW	1.8E-04	2.0E-07	2.2E-06	8.0E-07	6.0E-07	8.8E-06
%PCW-A	Loss of A Train PCW	1.6E-03	1.8E-06	1.98E-05	7.2E-06	5.4E-06	7.9E-05
%PCW-B	Loss of B Train PCW	1.6E-03	1.8E-06	2.0E-05	7.2E-06	5.4E-06	7.9E-05
%IA	Loss of All Instrument Air	6.5E-03	7.2E-06	7.9E-05	2.9E-05	2.2E-05	3.2E-04
%SDC (1 Trn)	Loss of Shutdown Cooling (Both train)	2.6E-01	2.6E-04	N/A	N/A	N/A	N/A
%SDC (2 Trns)	Loss of Shutdown Cooling (Both train)	7.54E-03	N/A	8.3E-05	3.0E-05	2.3E-05	1.8E-07

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

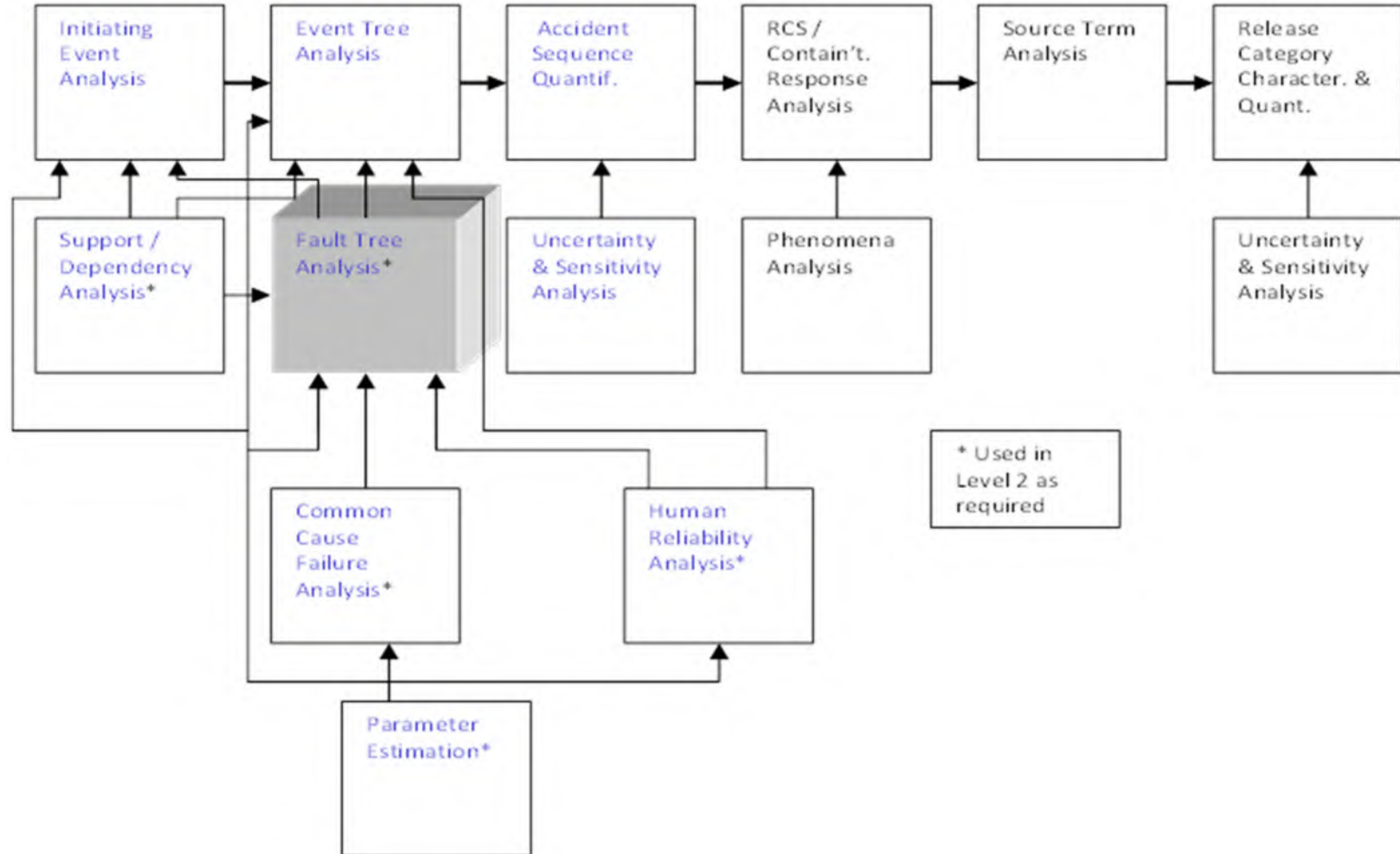
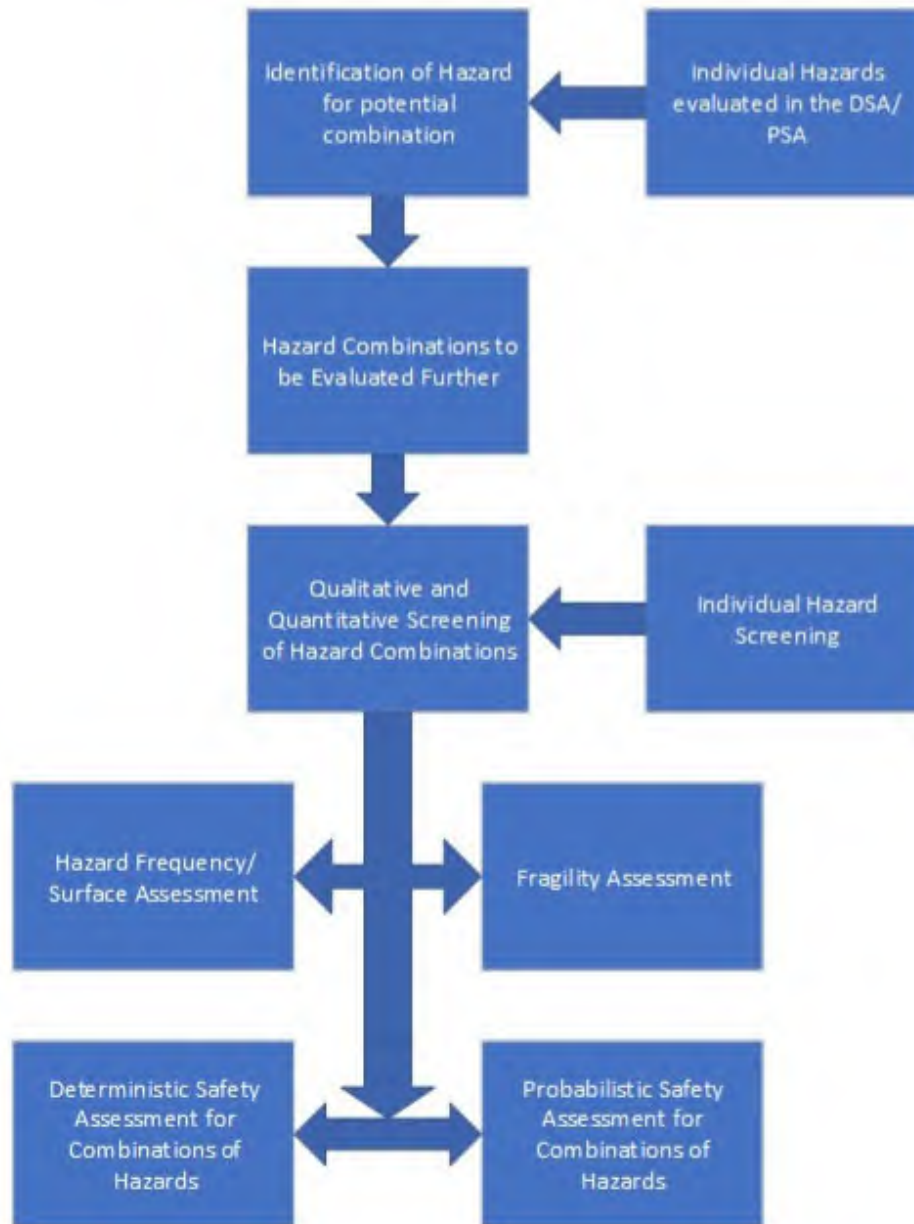


Figure 15.6-1: Principal Steps in Probabilistic Safety Assessment



**Figure 15.6-2: Overall Process for Selection, Screening and Analysis of Hazards**

## **15.7 Results of the Deterministic Safety Analyses and Probabilistic Safety Assessment**

The DSA results for the bounding BWRX-300 events meet acceptance criteria in Tables 15.3-2 and 15.3-3, respectively. Results Summary for the DSA is provided in the following tables:

Table 15.7.2-1	Results Summary of the AOO Safety Analyses
Table 15.7.3-1	Results Summary of DBA and DEC Events for Non-LOCA
Table 15.7.3-2	Results Summary of the Fuel Loading Error Event
Table 15.7.3-3	Results Summary of DBA and DEC Events for LOCA
Table 15.7.6-1	Fuel Handling Accident Dose Consequences
Table 15.7.7-1	BWRX-300 DNNP Main Steam Line Break Outside Containment – Accident Dose Consequences
Table 15.7.7-2	BWRX-300 DNNP Feedwater Line Break Outside Containment - Accident Dose Consequences
Table 15.7.7-3	BWRX-300 DNNP Isolation Condenser System Line Break Outside Containment – Accident Dose Consequences
Table 15.7.7-4	BWRX-300 DNNP Instrument Line Break Outside Containment Accident Dose Consequences

The results for the beyond design basis events evaluated in the PSA are provided in Table 15.7.9-1 “Comparison of PSA Results with CNSC REGDOC-2.5.2 Limits” indicate that the design does not exceed the safety goals in Table 15.3-3.

Implementation of the D-in-D concept ensures multiple, independent layers of protection against unacceptable radiation releases. None of the bounding AOOs, DBAs or DEC Events Without Core Damage analyzed approach the regulatory limits for radioactive releases.

### **15.7.1 Results of Analysis of Normal Operation**

The analysis of normal operations for stability is described in Chapter 4, Section 4.8 and Section 15.5.2.4. Subsection 15.5.2 describes the normal operations DSA. The OLC (Chapter 16) provides the operating limits and parameters ensuring the plant is operating within the design basis.

### **15.7.2 Results of Analysis of Anticipated Operational Occurrences**

The analysis of AOOs is described in Subsection 15.5.3. The resulting maximum neutron flux, dome pressure, RPV bottom pressure, simulated thermal power, and  $\Delta$ CPR/ICPR for AOOs is provided in Table 15.7.2-1.

### **15.7.3 Results of Analysis of Design Basis Accidents**

The analysis of DBAs is described in Subsection 15.5.4. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power (%NBR), and peak clad temperature from non-LOCA DBAs are provided in Table 15.7.3-1. The LOCA DBA summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.7.3-3.

### **15.7.4 Results of Analysis of Design Extension Conditions Without Core Damage**

The analysis description of DEC events without core melting is described in Section 15.5.5. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom

pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DEC are provided in Table 15.7.3-1.

The LOCA DEC summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.7.3-3. The summary results of a fuel loading error event DEC are provided in Table 15.7.3-2.

#### **15.7.5 Results of Analysis of Design Extension Conditions with Core Damage**

The analysis description of those DEC events associated with core damage are described in Section 15.6.3 Level 2 PSA.

#### **15.7.6 Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool**

The analysis description of those postulated events and accident scenarios associated with the fuel pool are described in Subsection 15.5.7.

#### **15.7.7 Results of Analysis of Fuel Handling Events**

The analysis of FHA is described in Subsection 15.5.8. The resultant dose consequences from an FHA is provided in Table 15.7.6-1.

#### **15.7.8 Results of Analysis of Radioactive Releases from a Subsystem or a Component**

The analysis description of the liquid and gaseous radioactive waste system process scenario releases from a postulated liquid tank failure or off-gas system failure are described in Chapter 11, Subsections 11.2.9, and 11.3.14, respectively. The off-gas system failure results are provided in Table 11.3-4 and meet the CNSC allowable limits.

#### **15.7.9 Results of Analysis of LOCA Breaks Outside Containment**

LOCA from pipe breaks outside containment described in Subsection 15.5.9.2 result in normal coolant concentration releases that are within allowable limits.

##### **15.7.9.1 Results of Main Steam Line Break Outside Containment**

The analysis of a MSL break outside containment is described in Subsection 15.5.9.2.1 and the results are provided in Table 15.7.7-1.

##### **15.7.9.2 Results of Large Feedwater Pipe Break Outside Containment**

The analysis of a large feedwater pipe break outside containment is described in Subsection 15.5.9.2.2, and the results are provided in Table 15.7.7-2.

##### **15.7.9.3 Results of Large Isolation Condenser Pipe Break Outside Containment**

The analysis of a large ICS pipe breaks outside containment is described in Subsection 15.5.9.2.3 and the results are provided in Table 15.7.7-3.

##### **15.7.9.4 Results of Small Breaks Outside Containment**

The analysis of a small pipe break outside containment is described in Subsection 15.5.9.2.4 and the results are provided in Table 15.7.7-4.

#### **15.7.10 Results of Analysis of Internal and External Hazards**

The description of internal and external hazards and associated design basis SSCs mitigating these hazards are described in Section 15.5.10 for the DSA.

#### **15.7.11 Results of Probabilistic Safety Analysis**

The Level 1 and Level 2 PSA are described in Subsection 15.6.2. The general PSA approach and the insights and applications are described in Subsections 15.6.1 and 15.6.3. The comparison of PSA results with CNSC REGDOC-2.5.2 acceptance criteria is shown in Table 15.7.9-1.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.7.2-1: Results Summary of AOO Events**

Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	$\Delta$ CPR/ICPR
Decrease in Core Coolant Temperature AOOs						
LFWH AOO (Section 15.5.3.1.1)	MOC	101.1	7.17 (1040.1)	7.33 (1062.6)	100.4	0.0341
Pressure Increase and Inventory Reduction AOOs						
LR-TT AOO (Section 15.5.3.2.1)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
1MSRIVC AOO (Section 15.5.3.2.2)	EOR	110.7	7.47 (1083.2)	7.61 (1103.1)	100.3	0.0631
LOCV AOO (Section 15.5.3.2.3)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
LOPP AOO (Section 15.5.3.2.4)	BOC	100.0	7.55 (1094.6)	7.70 (1116.2)	100.0	0.0495
FWPT AOO (Section 15.5.3.3.1)	EOR	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	0.0086
Increase in Reactor Coolant Inventory AOOs						
Inadvertent Isolation Condenser Initiation - One Train (Section 15.5.3.4.1)	MOC	116.8	7.17 (1039.7)	7.32 (1062.1)	100.6	0.0464



**Table 15.7.3-1: Results Summary of the DBA and DEC Events – Non- LOCA**

Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	PCT °C (°F)
<b>Decrease in Reactor Coolant Temperature Event DBA</b>						
LFWH DBA (Section 15.5.4.1.1)	MOC	119.1	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	115.2	308.0 (586.3)
<b>Increase in Reactor Pressure Events DBAs</b>						
LR DBA Note 2 (Section 15.5.4.2.1)	MOC	544.3	8.69 (1259.8)	8.86 (1285.1)	111.8	511.8 (953.3)
LOPP DBA (Section 15.5.4.2.2)	EOR	151.0	8.61 (1249.2)	8.73 (1266.1)	103.5	312.4 (594.3)
RPCD DBA (Section 15.5.4.2.3)	EOR	151.4	8.70 (1262.3) <sup>1</sup>	8.89 (1288.9) <sup>1</sup>	103.5	312.6 (594.7)
MSRIVC-FWIV DBA (Section 15.5.4.2.4)	EOR	158.9	8.61 (1249.4)	8.73 (1266.3)	103.7	312.7 (594.9)
<b>Increase in Reactor Coolant Inventory DBAs</b>						
FWFI – All Pumps (Section 15.5.4.4.1)	MOC	123.3	7.93 (1150.0) (Note 1)	8.09 (1173.6) (Note 1)	115.5	315.5 (599.9)
Inadvertent Isolation Condenser Initiation - All Trains (Section 15.5.4.4.2)	MOC	114.3	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)
<b>Decrease in Reactor Coolant Inventory DBAs</b>						
LOFW DBA (Section 15.5.4.5.1)	MOC	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	PCT °C (°F)
RPCO DBA (Section 15.5.4.5.2)	MOC	100.0	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	100.0	302.8 (577.0)
<b>Analysis of Design Extension Conditions Without Core Damage</b>						
<b>Pressure Increase DEC's</b>						
1MSRIVC DEC (Section 15.5.5.2)	EOR	208.9	11.19 (1623.2)	11.33 (1643.3)	130.1	727.1 (1340.8)
Complex Sequence LR DEC (Section 15.5.5.3)	EOR	118.4	8.03 (1164.3)	8.17 (1184.5)	100.3	309.2 (588.6)
LOCV DEC (Section 15.5.5.4)	EOR	245.0	9.98 (1447.3)	10.12 (1467.5)	124.5	332.9 (631.2)
LOPP DEC (Section 15.5.5.5)	EOR	250.1	11.14 (1615.6)	11.28 (1635.5)	130.4	328.9 (624.0)
<b>Reactivity and Power Distribution Anomalies - DEC's</b>						
CCF ACRW (Section 15.5.6.1)	MOC	123.8	7.52 (1091.0)	7.68 (1113.3)	115.4	314.0 (597.2)
ICRW (Section 15.5.6.2)	BOC	111.8	7.23 (1048.5)	7.38 (1070.9)	111.8	307.5 (585.5)
<b>Decrease in Reactor Coolant Inventory - DEC</b>						
FW Isolation DEC (Section 15.5.5.4)	MOC	100.0	8.40 (1218.5)	8.50 (1232.1)	100.0	308.0 (586.3)

(1) The simulation of this event is ended before the RPV pressure increase resulted in an ICS train initiation. Assuming a single failure, the second ICS train initiates at 8.605 MPa (1248 psia) and decreases pressure. The pressure remains well below the acceptance criteria for the event. Results are from a bounding case with combined conservatisms.

(2) Results are from a bounding case with combined conservatisms.

**Table 15.7.3-2: Results Summary of Fuel Loading Error Event**

<b>Dispersion Factor 0-2 hour (s/m<sup>3</sup>)</b>	<b>Whole-Body Dose</b>
3.00E-04	0.095-mSv (9.5 mrem)

**Table 15.7.3-3: Results Summary of DBA and DEC Events For LOCA**

<b>Parameter</b>	<b>Value</b>
<b>Results for Main Steam Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.1)</b>	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	423 kPa (0.322MPaG)
Peak containment shell temperature	134 °C
<b>Results for Feedwater Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.2)</b>	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	407 kPa (0.306 MPaG)
Peak containment shell temperature	134 °C
<b>Results for Small Steam Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.4)</b>	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa (0.090 MPaG)
Peak containment shell temperature	125 °C
<b>Results for Small Liquid Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.4)</b>	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa (0.090 MPaG)
Peak containment shell temperature	110 °C

**Table 15.7.6-1: Fuel Handling Accident (DEC) Dose Consequences**

Dose Location	30-Day Whole-Body Dose <sup>Note 1</sup>	Acceptance Criteria
350 m EZ	2.91 mSv (0.291 rem)	20 mSv (whole-body dose for a 30-day period)

(1) Maximum dose for the most critical group.

**Table 15.7.7-1: BWRX-300 DNNP Main Steam Line Break Outside Containment  
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	8.16E-01 mSv (8.16E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	2.7 mSv (2.7E-01 rem)	

**Table 15.7.7-2: BWRX-300 DNNP Feedwater Line Break Outside Containment  
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	9.29E-01 mSv (9.29E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	3.03 mSv (3.03E-01 rem)	

**Table 15.7.7-3: BWRX-300 DNNP Isolation Condenser System Line Break Outside  
Containment Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	2.63E-04 mSv (2.63E-05 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	1.18E-03 mSv (1.18E-04 rem)	

- (1) The dose consequences of an ICS line break at the BWRX-300 Darlington exclusion zone are very low. This is expected because the ICS break flow releases only steam that is assumed to entrain 2 to 3 orders of magnitude less activity than reactor water, and the ICS line break is isolated in a very short 10 second period.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.7.7-4: BWRX-300 DNNP Instrument Line Break Outside Containment  
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	9.99E-01 mSv (9.99E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	3.50E+0 mSv (3.50E-01 rem)	

**Table 15.7.9-1: Core Damage Frequency Results**

PSA Events	Core Damage Frequency (CDF) (/yr) <small>Note 1</small>
At Power	1.1 E-08
Internal Events and Low Power Shutdown (LPSD)	7.0E-10
Internal Fire	1.3E-08
Internal Flood	1.5.E-09
Seismic	5.1E-08
High Wind	4.3E-09 (Straight Wind) 1.3E-10 (Tornado)
Fuel Pool Events	1.3E-08
Fuel and Heavy Load Movements (heavy load drop can cause fuel damage or core damage)	FDF at Power 2.3E-09 FDF at LPSD 1.8E-09 CDF at LPSD 1.6E-09

(1) CNSC REGDOC-2.5.2 CDF limit is 1E-05.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.7.9-2: Large Release Frequency Results**

Level 2 PSA Scope	Large Release Frequency (LRF) Events per Year <sup>Note 1</sup>	Comment
Internal Events At Power	1.8E-09	LRF consists of large early and large late release frequency. SRF is not reported because a small release is conservatively assumed to progress into a large release.
Internal Events Low Power & Shutdown	7.0E-10	LRF is same as CDF because containment is conservatively assumed open during low power and shutdown.
Seismic Events	4.8E-08	With seismic hazard truncation.
Fire Events	1.3E-8	LRF is conservatively assumed the same as CDF. CDF is 1.3E-08/yr.
High Wind	4.3E-09/yr. for straight wind 1.3E-10/yr. for tornado	LRF is conservatively assumed the same as CDF.
Internal Flood	5.5E-10	
Fuel and Heavy Load Movements	5.7E-09	
Fuel Pool Events	1.3E-8	

(1) CNSC REGDOC-2.5.2 LRF is 1E-06.

## **Appendix 15A. Reference Source Term for Conditions That Are Practically Eliminated**

A radiological and combustible gas accident reference source term based on a set of representative core damage conditions is established for use in specifying design attributes and/or complementary design features for DECAs as required by Section 7.3.4 of CNSC REGDOC-2.5.2.

### **15A.1 Safety Objectives**

The safety objectives are consistent with CNSC REGDOC-2.5.2, Section 7.3.4 (Chapter 3, Reference 3.1-1) and IAEA SSR-2/1 (Chapter 3, Reference 3.1-5). The BWRX-300 design fault sequences that could lead to early or large radioactive releases are practically eliminated.

### **15A.2 Acceptance Criteria**

Fault sequences with early or large releases are practically eliminated if either of the following criteria are met:

1. It is physically impossible for the relevant fault sequences to occur
2. There is a high level of confidence that these fault sequences are extremely unlikely

### **15A.3 Approach**

Consistent with Chapter 3, Section 3.1.8, the identification of DECAs is commenced early in the design process, with iterations between design activities, DSA, and PSA with an increase in scope and level of detail as the design progresses.

As part of the overall strategy for addressing challenges, plant conditions, and for restoring and maintaining FSFs, the roles of each complementary design feature and other design attributes are defined. The objective is demonstrating that these complementary design features design attributes are reasonably expected to practically eliminate significant radioactive releases for DECAs.

Each complementary design feature or plant attribute is defined as part of the overall plant strategy for addressing challenges, plant conditions, and for restoring and maintaining the FSFs. The objective is demonstrating that the planned operation of these complementary design features or design attributes is reasonably expected to practically eliminate significant radioactive releases for DECAs:

1. The physical conditions, processes, and phenomena associated with the accident progression.
2. Risk significant severe accident scenarios involving hazards selected for analysis.
3. Bounding fault sequences and DSA considered in the plant design envelope and associated attributes.

The outcome is a set of conditions that are considered in the design of corresponding complementary design safety features or design attributes for DECAs and represent a set of bounding conditions that envelope other accidents.

DECAs selected are justified based on engineering judgement, DSA results and insights from the PSA. In addition, the practical elimination concept reinforces D-in-D by providing a focused evaluation of those conditions having the potential for early radioactive release or a large radioactive release.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The initial DEC selection of is based on predecessor BWR reactor research and OPEX for the purpose of framing the conceptual design. A more rigorous selection is achieved with iterations between design activities, DSA, and PSA with an increase in scope and level of detail as the design progresses. The reference source terms for the selected conditions are used to develop and confirm the complementary design features or design attributes

If the accident sequence is practically eliminated as a result of a single initiating event such as the failure of the RPV in all operational states, the demonstration relies on achieving a high level of quality at all stages of the component lifetime.

A condition is practically eliminated by considering:

- DL features consisting of equipment and supporting operational organizational provisions (e.g., SAMGs). Certain equipment types may receive increased DL1 design provisions such as providing highly reliable components that are shown not to fail as part of the DSA.
- DL strength – adequate margins, reliability, and qualification against all operating conditions
- DL independence – redundancy physical separation, diversity, and functionality

#### **15A.4 Practically Eliminated Events**

For the conceptual design, practical elimination considerations are essential in framing conditions where complementary design features or design attributes are used with the accompanying rationale for selecting DEC's that comprise the reference source term.

The following practically eliminated conditions and provisions achieving practical elimination in Table 15A-1 are based on deterministic and probabilistic methods, operational experience, engineering judgement, and the results of research and analysis.

**Table 15A-1: Practically Eliminated Conditions**

<b>Practically Eliminated Conditions</b>	<b>Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions</b>
A sudden mechanical failure of the RPV that eliminates the capability to hold and cool the core.	<p>DL1 provisions for a strong reactor coolant pressure boundary for all states within the plant design envelope without relying on subsequent DLs, including DEC. The following selection of RPV materials and operational uses throughout the plant lifetime provides greater confidence that the likelihood of sudden RPV failure is extremely unlikely:</p> <ul style="list-style-type: none"> <li>• Material fracture toughness</li> <li>• Tensile properties</li> <li>• Defect free fabrication</li> <li>• Defect tolerance</li> <li>• Structural integrity</li> <li>• Operational limits for all operational conditions (normal, AOOs, DBAs)</li> <li>• Loading conditions</li> <li>• Highest reliability and quality according to ASME BPVC</li> </ul>
A sudden mechanical failure of the RIVs that results in catastrophic failure creating significant early or late containment challenges.	<p>DL1 requirements for the highest reliability and quality according to ASME BPVC Code Class 1 components provides greater confidence that the likelihood of sudden mechanical failure of the RIVs (part of the RCPB) throughout the plant lifetime is extremely unlikely.</p> <p>These requirements provide assurance that the quality, materials of construction, fabrication, installation, testing, examination, repair/replacement, and ongoing inspection, testing, and monitoring are commensurate with the RIV FSF.</p> <p>DL1 design provisions for nozzle elevations significantly above the core, and passive isolation condenser cooling minimize the likelihood of substantial core damage in the event of a coolant loss from RIV failure.</p>
A severe and unlikely condition where more than 50% of the control rods fail to insert leading to an accident with increasing severity that challenges the reactor and containment system is practically eliminated.	<p>The design offers D-in-D with diverse and redundant shutdown systems making multiple sequential failures extremely unlikely. However, conservatively, the design includes complementary design features that provides a diverse means limiting increases in the reactor vessel pressure in the extremely unlikely condition where more than 50% of the control rods fail to insert.</p> <p>DL4a complementary design feature diverse protection system actuation logic that is diverse and independent from the DL3 SC1 I&amp;C System actuation logic that provides greater confidence that the condition is extremely unlikely.</p> <p>DL4a complementary design feature of the FMCRD run-in function provides greater confidence that the condition is extremely unlikely.</p>

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
	<p>DL4a complementary design feature of alternate control rod insertion ensures the hydraulic control unit scram pressure is released to the FMCRDs even in the unlikely event of the solenoid pilot valves failing to scram. This provides greater confidence that the condition is extremely unlikely.</p> <p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit increases in reactor vessel pressure for this extremely unlikely condition, thus making failure propagation from the reactor pressure vessel to the containment extremely unlikely.</p> <p>DL4b BIS (see Appendix 15B.2) complementary design feature provides a means to place the plant in a stable controlled configuration if control rods fail to insert.</p>
<p>A single control rod falling out of the core and inserting enough positive reactivity to cause a calculated fuel enthalpy rise greater than the lower bound clad failure limits.</p>	<p>DL1 design provision that the control blades include a bayonet style coupling design provision that requires a 45-degree rotation to uncouple, thus making it physically impossible for the control rod blade to become uncoupled from the drive during reactor operation.</p> <p>DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the FMCRD and the control rod, or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur.</p> <p>Chapter 4, Section 4.5 discusses these design provisions for the FMCRD and the control rods.</p>

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
<p>Containment heating following a postulated RPV failure does not occur.</p> <p>This chain of events is called a high pressure melt ejection and these four mechanisms may cause a rapid increase in containment pressure and temperature:</p> <ol style="list-style-type: none"> <li>1. Blowdown of the reactor pressure vessel</li> <li>2. Efficient debris-to-gas heat transfer</li> <li>3. Exothermic metal/steam and metal/oxygen reactions</li> <li>4. Hydrogen combustion</li> </ol> <p>These mechanisms lead to increased loads on the containment building that are collectively referred to as direct containment heating.</p>	<p>DL1 design reduces the likelihood of being at a high pressure at the time of vessel failure is eliminated by the large inventory of coolant in the RPV. Core cooling challenges are only possible when corresponding RPV depressurization occurs. With adequate control rod insertion, RPV depressurization is eliminated by the DL3 passive ICS and the DL3 RIV isolation or diverse DL4a RIV isolation.</p> <p>DL4a ICS actuation provides greater confidence that RPV failure occurring at high pressure is extremely unlikely.</p> <p>DL4b RPV ultimate pressure regulation complementary design feature provides a diverse means to limit RPV pressure increases ensuring that there are no credible core damage accident sequences without cooling when the reactor is pressurized, making direct containment heating following a postulated vessel failure highly unlikely with a high degree of confidence.</p> <p>DL4b complementary design feature is the containment ultimate overpressure vent that ensures containment does not overpressurize when RPV ultimate pressure regulation is actuated.</p>
<p>Containment challenges from large steam explosions do not occur.</p> <p>Molten core debris interacting with water can produce explosive interactions as well as non-explosive, dynamic interactions from mutual fragmentation and mixing of the core material and water.</p>	<p>RPV pressure is sufficiently high to suppress explosive conditions where anticipated thermal interactions are dynamic but not explosive.</p> <p>DL1 design provides a strong containment structure and components and sufficiently large containment volume making containment load challenges from large steam explosions physically impossible.</p> <p>DL1 design and operating physical limitations (e.g., no subcooled water pool) associated with explosive interactions preventing containment challenges.</p>
<p>Containment integrity challenges from combustible gas detonation does not occur.</p>	<p>DL1 design provision for an inert containment atmosphere and combustible gas management operating provisions limit concentrations well below detonation levels. This makes containment challenges due to combustible gas detonation physically impossible.</p> <p>DL4b combustible gas mitigation devices (e.g., passive mixing and venting, passive autocatalytic recombiners) are strategically located above the containment head in the vicinity of the pools. This provides greater confidence that any combustible gas leakage from containment does not result in damage to SSCs required for safety.</p>

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
Containment failure because of molten core concrete interaction.	If the RPV fails, a DL1 design provision for a corium shield/liner confines and prevents contact between the molten core and the containment liner. The corium shield/liner spreading area is cooled by the over-lying water. Steam is released through the hardened vent preventing containment failure.
Containment failure by quasi-static overpressurization from a long-term loss of containment heat removal (SA with unsuccessful containment isolation and passive containment heat removal).	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a long-term loss of containment heat removal physically impossible.
An interfacing system LOCA outside containment.	DL1 design provisions ensures subsystems connected to the RCPB are designed to an ultimate rupture strength at least equal to the reactor coolant system design pressure make an interfacing system LOCA physically impossible.
Containment bypass consequential to SA progression. SA fault sequences leading to or involving RIV leakage and core damage.	DL1 RIV design includes ball valves with hydraulic actuators. Design, qualification, and valve procurement ensures the important FSF of RPV isolation occurs.  DL1 design and operating provisions for a strong RCPB for all states within the plant design envelope are the highest reliability and quality complying with ASME BPVC. The mechanical failure likelihood of the RIV internals throughout the plant lifetime due to SA conditions is extremely unlikely.
A SA with an open containment (plant in shutdown state). SA events during shutdown involve loss of fuel pool cooling sequences or LOCAs leading to drain-down.	DL1 design provisions of a large inventory of available water and operating provisions with slow developing fault sequences provide ample time for several diverse means of providing makeup water and preventing fuel being uncovered in the fuel pool or the reactor core. These design features provide greater confidence that the likelihood of SA with an open containment is extremely unlikely.
Fuel coolant interaction (in-vessel steam explosion).	DL1 design provisions of a passive ICS (ECCS system) with a large water inventory with inherent margins eliminates system challenges leading to loss of adequate core cooling. This provides confidence that the likelihood of an in-vessel steam explosion from loss of core cooling is extremely unlikely.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
A high pressure melt ejection with potential for debris dispersion such that debris is not retained in lower containment.	<p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit the increase in reactor vessel pressure ensuring that there are no credible core damaging accident sequences without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection physically impossible</p> <p>DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a high pressure melt ejection physically impossible.</p> <p>DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction physically impossible.</p>
An out of core criticality event during fuel handling.	<p>DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction, and spacing.</p> <p>Plant procedures prohibit the handling and storage of more than one fuel assembly. The amount of water moderator available from internal flooding is limited by the design of the storage location structures thus making an out of core criticality physically impossible. Refer to Section 15.5.9.3 for the BWRX-300 Out of Core Criticality Analysis.</p>

## **Appendix 15B. Complementary Design Features for Mitigating Design Extension Conditions**

Consistent with CNSC REGDOC-2.5.2, Section 6.1, “Application of a D-in-D,” and Section 6.3 “Accident prevention and plant safety characteristics,” complementary design features are used to prevent accident progression or mitigate the consequences of DEC. In accordance with the BWRX-300 Safety Strategy, complementary design features are identified that address plant challenges, conditions that require restoration, and maintaining the FSFs. These complementary design features can be reasonably expected to practically eliminate or mitigate significant radioactive releases for DEC. Practical elimination of DEC is discussed in the previous Appendix 15A. There are several complementary design features provided ensuring that DEC are either practically eliminated or extremely unlikely to occur. Complementary design features mitigating functions are provided in Table 15B-2.

Most complementary design features are component features such as vents and catalytic recombiners. However, the boron injection system described in the following section is a complete system used in inserting negative reactivity in the core in the highly improbable event where shutdown using CRDs via hydraulic or motor run-in cannot be accomplished.

### **15B.1 Boron Injection System**

The Boron Injection System (BIS) provides a separate, diverse means, D-in-D backup system to for manually inserting negative reactivity into the reactor core. The BIS assures reactor shutdown by mixing a neutron absorber with the primary coolant.

The BIS is only required in the highly improbable event when shutdown using CRDs via hydraulic or motor run-in cannot be accomplished. The system introduces sufficient negative reactivity into the reactor primary system assuring reactor shutdown to a subcritical state with no control rod motion.

Injection of a neutron absorber solution for reactivity control is a DL4b, Safety Category 3 function.

The system achieves cold shutdown conditions with a predetermined mass of water in the reactor vessel using an enriched Boron-10 solution injected into the reactor vessel. The CRD system and the BIS do not share any components.

#### **15B.1.1 Safety Design Bases**

The BIS provides a means of achieving cold subcriticality by mixing a neutron absorber with the primary coolant. This condition creates a design requirement for reactivity control from full power to cold 20°C (68°F) subcritical state using an enriched Boron-10 solution.

The ICS for overpressure control and CRD/HCU/FMCRD for reactivity control address failure-to-scrum impacts or other reactivity events where boron injection provides an additional layer of defence.

The neutron absorber injection is manually initiated by the operator from either the MCR or SCR.

The minimum amount of neutron absorber required to shut down the reactor is calculated based upon the minimum natural boron concentration required for shutdown and the weight of the water in the RPV at normal level including the shutdown cooling loops at cold shutdown conditions. This ensures that total quantity of stored neutron absorber is sufficient to achieve the minimum sodium pentaborate concentration required for shutdown.

The BIS is a complimentary design feature that provides D-in-D for long-term reactivity control DEC.

The BIS system is designated as Safety Class 3.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The BIS can be operated in the event of a LOPP but does not perform any non-safety or safety class functions during off-normal conditions.

The BIS injection line penetrates containment and is directly connected to the reactor vessel through the ICS. This portion of the BIS through the ICS system that requires containment isolation is part of the RCPB and is classified as DL3, Quality Group A, Safety Category 1, and Seismic Category 1-A and 1-B. The ICS equipment is Safety Class 1.

### **15B.1.2 System and Equipment Functions**

The BIS simplified flow diagram of the boron injection mode is shown on Figure 15B-2. The BIS consists of a storage tank, test tank, injection pump, piping, valves, and instrumentation and controls necessary to prepare and inject the neutron absorbing solution into the reactor and for system testing. The air-operated injection valve and the Class 1 injection piping are part of the BIS. The BIS consists of a single, 100% equipment train located in the RB outside containment, tying into an ICS return line between the redundant actuation valves and the RIVs.

The neutron absorber is pumped from the storage tank through the injection valves into the reactor vessel by remote manual operation of the pump, storage tank outlet valve, and injection valve.

The BIS injects the neutron absorber solution into the reactor core compensating for various reactivity effects that could occur during plant operation.

The neutron absorber is an aqueous solution of decahydrate ( $\text{Na}_2\text{B}_{10}\text{O}_{16} \cdot 10\text{H}_2\text{O}$ ). Enriched sodium pentaborate solution is made by mixing granular enriched sodium pentaborate with water. The neutron absorber design concentration has a 4.4°C solution precipitation temperature. This low precipitation temperature eliminates the need for storage tank heating during normal standby conditions or heat tracing the instrument lines and pump suction piping.

The volume versus concentration limits is calculated accounting for normal reactor vessel water volume and water volume in the shutdown cooling piping. This neutron absorber solution quantity is the amount above the pump suction shutoff level in the storage tank. No credit is taken for the portion of the storage tank volume that cannot be injected.

### **15B.1.3 Component Description**

The BIS is a single 100% capacity system that includes:

- A single triplex injection pump capable of 2.27 m<sup>3</sup>/hr (10 gpm) pumping against the maximum reactor pressure of 12.41 MPaG (1800 psig)
- A test tank and associated piping and valves
- An injection line with containment isolation valves
- A demineralized water connection
- A service air connection
- System piping drains and collection subsystem

All equipment is located outside primary containment allowing access for testing and inspection activities during all plant operating conditions.

Operation in the injection mode requires BIS pump start and opening the storage tank outlet valve and injection valves. The storage tank outlet valve and injection valve are air-operated. The air-operated valves open on a system initiation signal, and close [fail as-is] on loss or removal of the pressurized air source or close on a manual valve closure signal.



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

The injection location is the condensate return line of the ICS "C" loop downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location requires a separate containment penetration for the BIS injection line. Overpressure protection of the BIS is provided by a relief valve located on the pump discharge piping. To avoid water hammer at the onset of BIS injection, the piping is fully-filled by a connection with the demineralized water system.

The testing subsystem consists of a test tank containing demineralized water and the associated valves and piping in parallel with the normal suction flow path from the storage tank. During the non-testing phase of system normal operation, the testing network is isolated from the balance of the system by shutoff valves. Periodic operational tests of the BIS are conducted at any time regardless of the state of reactor operation. The testing subsystem demonstrates:

- Pumping demineralized water at rated pressure and flow to and from the test tank
- Pumping demineralized water from the test tank into the reactor vessel against existing reactor pressure (this test may only be performed during reactor shutdown) to ensure a functioning flow path

A tank zero level shutoff prevents damage to pump from insufficient suction. The storage tank, test tank, pump and interconnecting piping and valves are in the Reactor Building on the -8.5 m Level in Room 1551 that is nearest to the ICS connection line. There is a single containment penetration for the BIS injection line. The injection line connects with the ICS "C" return line downstream of the two condensate return valves providing a flow path into the reactor vessel.

#### **15B.1.4 Materials**

System piping meets ASME BPVC Section III, Subsection NC from the storage tank outlet connection up to the air-operated injection valve. The system piping from and including the air-operated injection valve (AOV) to the connection with the ICS is ASME B&PVC Section III, Subsection NB. Containment isolation is provided by a single check valve inside containment for immediate isolation and the check valve and AOV for long-term containment isolation. All system piping is either 304 or 316 stainless steel.

All equipment not required for injection, such as system drains, test equipment, and service air and demineralized water are classified as Quality Group D and designed to ASME B31.1 standard and are 304 or 316 stainless steel.

The BIS meets Seismic Class 1-A for the pump suction piping, pump discharge piping and injection piping up to the connection with the ICS. The AOVs and pump are Seismic Category 1-B. The test tank and test loop piping, demineralized water piping, service air piping, and system drains after the isolation valves are non-seismic.

The storage tank meets ASME B&PVC Section III, Subsection NC standards with 304 or 316 stainless steel and contains a hatch, heater system, sparger, and connections for an outlet, overflow, vent, service air and demineralized water inlet, and associated instrumentation.

The injection pump meets ASME B&PVC Section III, Subsection NC standards and is either 304 or 316 stainless steel for all wetted parts.

All equipment in the BIS in contact with neutron absorbing solution is stainless steel for corrosion protection.

#### **15B.1.5 Interfaces with Other Equipment and Systems**

System interfaces for the boron injection system are shown in Table 15B-2 and Figure 15B-1.

### **15B.1.6 System and Equipment Operation**

The BIS has four operating modes:

- Standby
- System Injection
- Pump Test
- System Injection Test Mode

BIS functions are required during power operation or reactor startup.

### **15B.1.7 Instrumentation and Control**

The BIS has the following instrument and control features:

1. High and low storage tank solution temperature annunciates in the control room at temperatures outside the allowed range
2. High and low storage tank solution level is annunciated in the control room when the level is outside its normal allowed limit
3. Low storage tank level automatically shuts off the injection pump when the solution level in the storage tank is below the zero level

Either the storage tank or test tank discharge valve must be fully open for the pump to run.

The following describes key instrumentation features of the BIS:

1. Storage tank solution is measured by a level transmitter. MCR level indication and high/low level alarms provided in the MCR HSI display.
2. Pump discharge pressure is sensed by a single pressure transmitter that provides MCR pressure indication through the HSI display
3. Storage tank solution temperature is sensed by a temperature element providing MCR storage tank solution temperature through the HSI display
4. Pump flow is sensed by a single flow element and flow transmitter with flow indication in the MCR
5. Pump status indication (energized status) provided in the MCR HSI display
6. System status indication and annunciators through the MCR HSI display for system bypass or inoperable condition
7. Valve position indication through the MCR HSI display for the storage tank outlet valve and test tank outlet valve
8. Pump overload trip or power loss indication provided through the MCR HSI display

A single pressure gauge is locally mounted for pressure indication during the pump test mode and is visible from the test loop control valve aiding control valve positioning.

### **Main Control Room Operation**

Manual initiation, operation, and control of the BIS is from the MCR through the HSI display. The means for manual actuation and for monitoring shutdown status from the MCR meets CNSC REGDOC-2.5.2, Section 8.4.3, addressing monitoring and operator action of shutdown systems.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

BIS actuation is dual-action initiation through the HSI display in the MCR that guards against accidental or inadvertent actuation of the system ensuring intentional system actuation.

Controls for the BIS are located through the HSI display in the MCR facilitating system operation. System and component operating status, including any system bypasses, system out of service, or manual overrides is provided. Manual system initiation and shutdown is provided.

The following BIS displays, alarms and controls are provided in the MCR HSI display:

**System Annunciators**

- High and low alarms for storage tank solution temperature
- High and low alarms for storage tank solution level
- Manual or automatic system out-of-service condition

**System Indication**

- Storage tank solution level
- Pump discharge pressure
- Pump operation (on/off)
- Storage tank and test tank outlet valves position (open/closed)
- Storage tank solution temperature
- System manually out of service
- Pump overload trip or power loss
- Pump flow rate

**Local Panel Operation**

A local panel is provided for the following:

**Controls:**

- Heater controls

**Indications:**

- Status indications for the mixing heater
- Storage tank solution temperature
- Storage tank solution level

**Secondary Control Room**

The BIS is initiated from the SCR if the MCR is not operational.

**15B.1.8 Inspection, Testing and Maintenance**

The BIS is functionally tested to ensure the pump develops rated flow and discharge pressure and the flow path from the pump suction to the RPV without contaminating the reactor with neutron absorber solution during each planned outage or reactor shutdown for refueling. Functional testing is performed by circulating demineralized water from the test tank, return to the test tank for the flow and pressure test, and with the injection valve open for the RPV injection test.

### **Maintenance Provisions**

The BIS design is provided with adequate equipment removal paths and personnel access for repair and replacement. The following features are provided to facilitate component maintenance:

1. The relief valve is provided with flanged inlet and outlet connections facilitating removal for bench testing
2. Isolation valves are provided on the pump suction and discharge side for maintenance
3. The storage tank heater can be removed without draining the tank
4. Sufficient pull space is provided for removing/replacing the storage tank heater
5. The storage tank is fitted with a top-mounted entry hatch and an external ladder for equipment installation, maintenance, and chemical addition
6. In the event of a sodium pentaborate system injection or the sodium pentaborate leakage into the system piping, the demineralized water supply can be used to flush the piping downstream the storage tank outlet valve
7. Sufficient pump room is provided for removing plungers, pistons, rods, crankshaft, and inspecting parts
8. Ample head room is provided above the pump for a crane, hoist, or tackle
9. Sufficient space is provided around the storage tank for inspection and maintenance
10. Sufficient space is provided for access and local instrument calibration
11. System maintenance is performed by closing the storage tank outlet isolation valve
12. A maintenance valve downstream of the inboard containment isolation valve can be closed allowing system maintenance and isolation valve testing
13. Test vents and drains for testing CIVs

Dedicated piping and collection for sodium pentaborate solution is provided where drainage occurs. The collection vessel is a portable stainless steel drum. A low containment surrounds the system preventing the spread of sodium pentaborate leaks. No special temporary or permanent provisions such as ladders, scaffolding, overhead cranes, etc., are required for maintenance.

### **Surveillance Testing and In-Service Inspection Provisions**

All active BIS components can be tested during normal plant operation. Pre-operational tests are conducted demonstrating the system flow path, adequate pump net positive suction head (NPSH), and pressure drops in accordance with system design documents.

Pre-operational and periodic hydrostatic testing of the system per OLCs are performed complying with ASME B&PVC Section III. Periodic testing of the system and components is scheduled at a frequency commensurate with the OLCs and meets the requirements of CNSC REGDOC-2.5.2, Section 8.4.2.

#### **15B.1.9 Radiological Aspects**

There are no radiological releases from a failure in the boron injection system. Breaks outside containment for the injection flow path through the injection location in the condensate return line of the ICS "C" loop downstream of the two ICS condensate return valves through a separate containment penetration is evaluated in Chapter 15, Subsection 15.5.9.2.3.

#### **15B.1.10 Performance and Safety Evaluation**

The BIS injects a neutron absorber quantity into the reactor vessel producing a minimum concentration to achieve a cold shutdown accounting for leakage and imperfect mixing.

The required shutdown concentration is achieved in a mass of water equal to the sum of the mass of water in the reactor vessel plus the mass of water in the SDC system.

The boron injection rate is based on a boron concentration rate of change between 8 to 20 ppm/minute in the reactor water that includes the weight of water in the reactor and shutdown cooling loops at normal level and at 20°C (68°F). The BIS pump can inject the sodium pentaborate solution at all reactor pressures from 12.41 MPaG (1800 psig) (vessel bottom) to zero MPaG.

The BIS can be actuated and operated in the event of LOOP. BIS contains safety features such as containment isolation and reactor coolant pressure boundary piping (Class 1 piping) for the injection line. The BIS injection line is part of the ICS described in Chapter 6, Subsection 6.2.1.

**Table 15B-1: Complementary Design Features Mitigating Functions**

DEC Function	Complementary Design Feature
Reactivity Control	<p>DL4a complementary design feature diverse protection system actuation logic that is diverse and independent from the SC1 actuation logic.</p> <p>DL4a complementary design feature of the FMCRD run-in function.</p> <p>DL4a complementary design feature of alternate control rod insertion ensures the hydraulic control unit scram pressure is released to the FMCRDs even in the unlikely event of the solenoid pilot valves failing to scram.</p> <p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit increases in reactor vessel pressure for this extremely unlikely condition, thus making failure propagation from the reactor pressure vessel to the containment extremely unlikely.</p> <p>DL4b BIS complementary design feature (see Section 15B.1) provides a means to place the plant in a stable controlled configuration if control rods fail to insert.</p> <p>Ex-vessel sequences do not result in a critical configuration.</p>
RPV Depressurization	See reactivity control complementary design feature for RPV ultimate pressure regulation.
In-Vessel Core Cooling	ICS includes DL4a actuations ensuring core cooling availability.
Cooling of Corium Debris in the Containment Corium Shield/Liner	<p>The corium/shield liner complementary design feature prevents core melt debris from interacting with the containment liner preventing core-concrete interaction.</p> <p>Debris bed cooling is accomplished by flooding the corium/shield liner with a pool of water following RPV failure. Heat transfer from the corium debris bed to the containment water pool prevents containment temperature rising that occurs due to heat radiated from an uncovered debris bed. In this manner, potential high temperature challenges to mitigating equipment and containment pressure boundary components such as penetrations seals are limited.</p>
Cooling High Pressure Melt Ejection Debris	<p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit the increase in reactor vessel pressure ensuring that there are no credible core damaging accident sequences without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection physically impossible</p> <p>DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a high pressure melt ejection physically impossible.</p> <p>DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction physically impossible.</p>

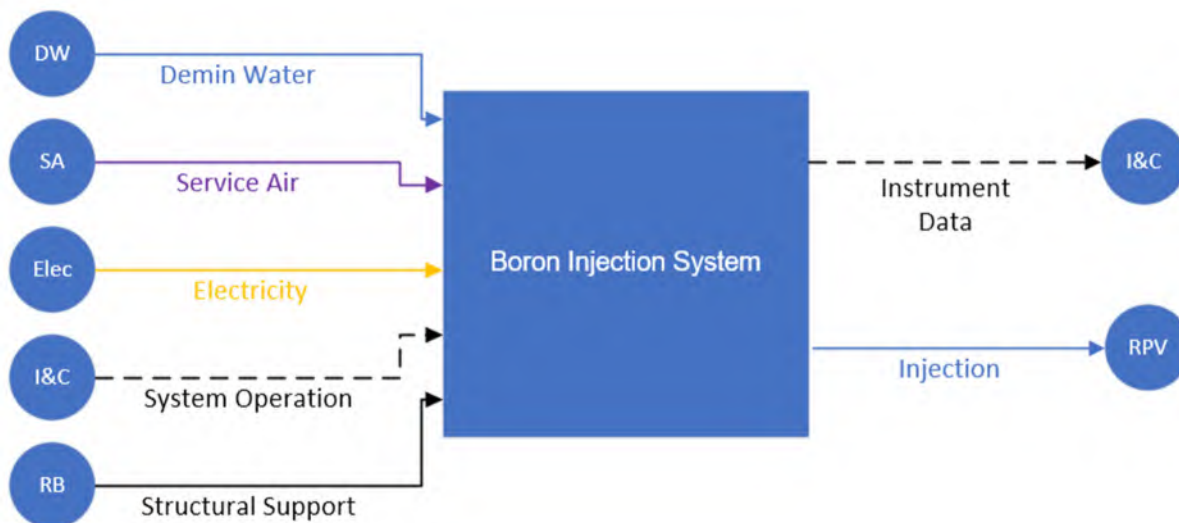
NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

DEC Function	Complementary Design Feature
Containment Isolation	<p>DL3 containment isolation is maintained providing a barrier to radionuclide releases to the environment.</p> <p>Diverse independent DL4a isolation functions are provided in the event of a CCF` of DL3 isolation functions. The CIV function includes the containment structure, penetration, valves, and isolation barriers.</p>
Containment Pressure Control:	<p>Containment pressure control via heat removal is required to prevent containment over-pressurization due to steam generation and/or exothermic chemical reactions (non-condensable gas generation).</p> <p>Passive containment cooling is normally in service including an SA event.</p> <p>A DL4b containment ultimate overpressure vent is included to relieve pressure to the reactor cavity and equipment pool to prevent potential containment failure.</p>
Combustible Gas Control	<p>Combustible gas control is present at the start of a potential accident sequence due to the inert containment atmosphere. Hydrogen generation could develop if a SA occurs. Oxygen during a SA is insufficient to create a combustible containment atmosphere.</p> <p>The ICS includes a non-condensable gas removal design feature (autocatalytic recombiner discussed in Section 6.2) that ensures combustible gas concentrations do not develop.</p>
Post-Accident Monitoring	<p>Post-accident monitoring provides information to facilitate post-accident response and evaluation of RPV and containment conditions.</p>

**Table 15B-2: Boron Injection System Interfaces**

<b>Interfacing System</b>	<b>Interface Description</b>	<b>Interface Boundary</b>
Water, Gas, and Chemical Pads	Provide demineralized water for initial storage tank fill and chemical mixing, system flushing, test tank fill for system testing, and maintaining the system full during standby.	The interface is at the upstream side of BIS pressure control valve.
Plant Pneumatics System	Provide service air for periodic storage tank solution mixing through the air sparger in the tank.	The interface is at the upstream side of BIS pressure control valve.
Safety Class 2 and 3 Electrical Distribution System	Electrical power for the BIS pump.	The interface boundary is at BIS equipment terminals.
Non-Safety Electrical Distribution System	Electrical power for the storage tank electric heater.	The interface boundary is at BIS equipment terminals.
Electrical Power	Provide electrical power for operation of the storage tank heater, and injection pump.	The interface boundary is at BIS equipment terminals.
Reactor Building Structure	Provide structural support and protection for BIS equipment.	The interface is at the equipment foundation and support structures for the BIS equipment.
Safety Class 2 and Safety Class 3 Instrumentation and Control	Provide for input and output for BIS instrumentation and controlled components.	The interface is at the DCIS.
Isolation Condenser System	Provide an injection path to the reactor for the neutron absorber solution and system isolation from the reactor system.	At the ICS "C" loop return piping.





**Figure 15B-1: BWRX-300 Sodium Pentaborate Injection System Interfaces**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

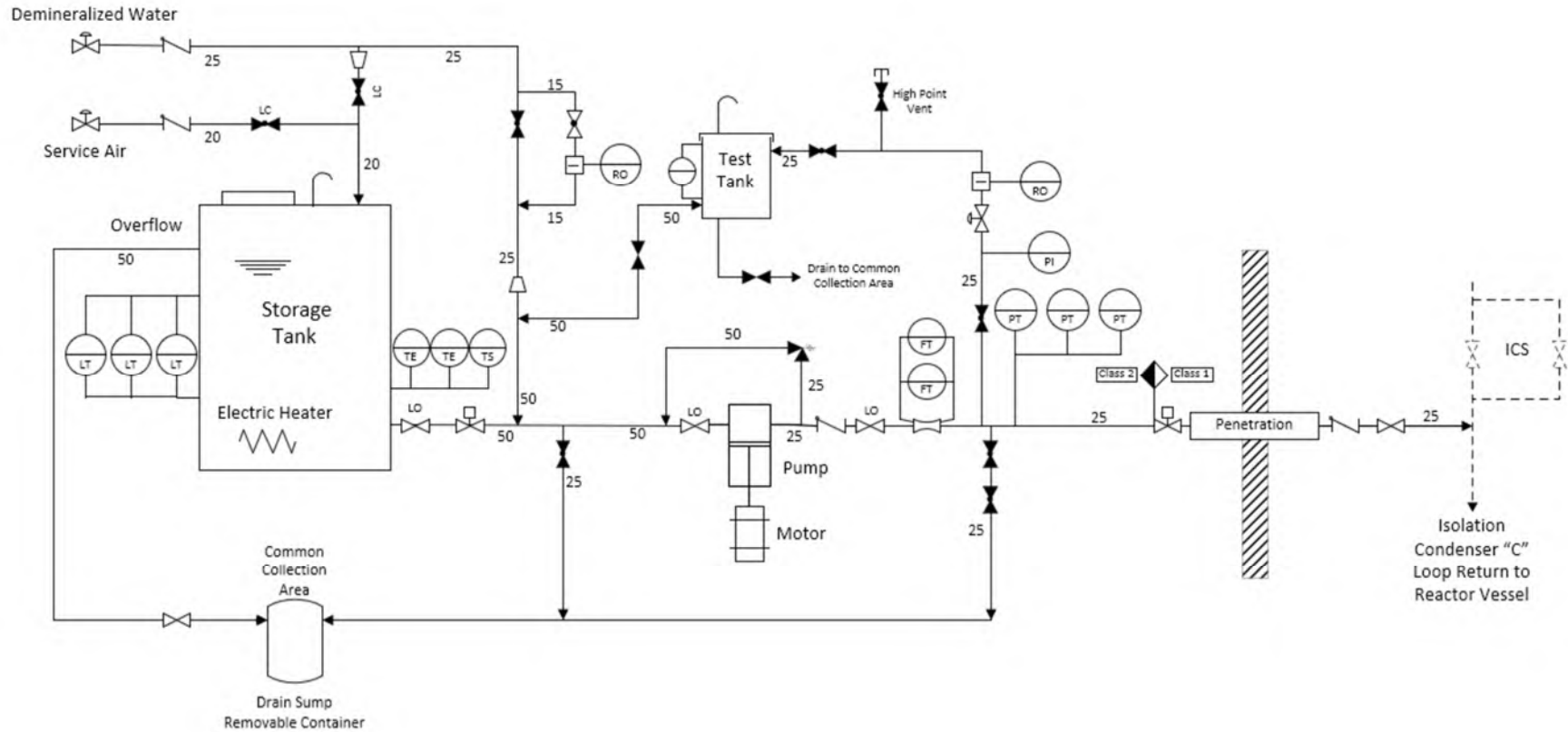


Figure 15B-2: Boron Injection System Simplified Flow Diagram

**Table 15.5-1: Key Initial Conservative LOCA Evaluations**

Parameter	Value	Notes
Thermal Power	887.4 MW	102% of rated power. Hot shutdown initial power is also included in steam pipe break cases
Dome pressure	7308.4 kPa	Upper end of normal operating range
Initial feedwater temperature	241.9 °C for MSL Break	Initial temperature of 191.9 °C is also included in FW pipe break cases
Initial temperature of the ICS and reactor cavity pools	43.3 °C	
Initial containment pressure	119.7 kPa	Upper end of normal operating range
Initial containment temperature	43.3 °C	Lower end of normal operating range
Initial water level in downcomer	21.1 m (large breaks)	Lower initial level (0.152 m below normal level) is also evaluated for small breaks

**Table 15.5-2: Summary of Core-Wide Decay Ratio Results**

Cycle Exposure (GWd/ST)	Core-Wide Decay Ratio	
	Nominal (241.9 °C)	LFWH to 191.9 °C with SCRR mitigation (AOO Conditions)
0 (BOC)	0.56	0.53
4 (MOC)	0.71	0.57
6 (EOR)	0.64	0.56

**Table 15.5-3: Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses**

Parameter	Value
<b>Heat Balance Related Parameters</b>	
Rated thermal power level, MWt	870
Core flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated core flow, kg/s (Mlbm/hr)	1890.0 (15.0)
Steam flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated steam flow, kg/s (Mlbm/hr)	503.2 (3.993)
Feedwater (FW) flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated FW flow, kg/s (Mlbm/hr)	507.0 (4.024)
Nominal dome pressure, MPa (psia)	7.17 (1040)
Nominal FW temperature, °C (°F)	241.8 (467.3)
Normal Reactor Water Level (NWL, L5), m (in) Above Vessel Zero (AVZ)	21.097 (830.591)
<b>Control Rod Drive / High Pressure Injection System Related Parameters</b>	
Control rod position versus time	Table 15.5-4
Control Rod Drive Mechanism (CRDM) run-in fast speed (minimum requirement), mm/s (in/s)	70.0 (2.76)
CRDM withdrawal (maximum speed), mm/s (in/s)	28.0 (1.10)
<b>Main Steam Line (MSL) Related Parameters</b>	
Number of MSLs	2
Minimum MSL length (average of all lines): flow path from vessel to Turbine Stop Valve (TSV), m (ft)	59.0 (194)
Minimum MSL volume (total of all lines including header): vessel to TSV, cubic m (cubic ft)	25.07 (885.3)
Minimum MSL pressure difference between the vessel dome pressure and the turbine throttle pressure at rated conditions, kPa (psi)	207 (30.0)
MSRIV minimum closure time, s	3.0
MSRIV closure profile for minimum closure time, s	
100% open area	0.0
100% open area	0.6
1% open area	1.7
0% open area	3.0
<b>Feedwater Related Parameters</b>	
Number of FW pumps	2
Number of motor-driven FW pumps	2
One FW pump is operating, and one FW pump is in standby	

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Parameter	Value
Maximum flow demanded is between, % Rated	105 - 115
Maximum flow of both FW pumps assuming a CCF, % Rated	220
FW temperature reduction for Loss of Feedwater Heating (LFWH) AOO, °C (°F)	50 (90)
Minimum time constant of FW temperature response, s This value is assumed based on operating experience and allows simplified modeling of the FW temperature response	60
FW pump trip coast down time constant	3
<b>Isolation Condenser System (ICS) Related Parameters</b>	
ICS heat removal capacity per train, MW	33.75
ICS maximum initial temperature, °C (°F)	43.3 (110)
ICS nominal initial temperature, °C (°F) (used in most transient events as the value is not key)	20.85 (69.53)
ICS minimum initial temperature, °C (°F)	10 (50)
ICS condensate return valve maximum opening time, s	10
ICS train condensate return line minimum internal diameter, mm (in)	178 (7.00)
The elevation difference from centerline of the horizontal connections between the ICS steam line distribution header to the centerline of the ICS return injection to the chimney, minimum, m (in)	15.808 (622.362)
<b>Balance of Plant Related Parameters</b>	
TSV minimum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV minimum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV minimum slow (servo) closure steam flow shutoff from rated power % rated steam flow / s	40
TBV capacity at rated conditions % rated steam flow	25
<b>Miscellaneous Parameters</b>	
Biased initial CPR, (reduction versus nominal initial CPR) Note: the value is approximate because it is slightly different depending on the initial condition. This is used in CN-PA events and some DEC events (as noted in event description)	~0.1
Biased initial hot fuel rod power, (multiplier relative to average rod power). Note: This is used in CN-PA events and some DEC events (as noted in event description). This input is only important if there is boiling transition and a subsequent fuel cladding temperature increase during the event.	1.6

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-4: CRD Scram Time**

Reactor Vessel Bottom Gauge Pressure MPaG (psig)	Rod Insertion Position	Required Maximum Time (s) Note 1
≤ 8.75 MPaG (1269 psig)	10%	≤ 0.46
	40%	≤ 1.20
	60%	≤ 1.71
	100%	≤ 3.70
≤ 9.48 MPaG (1375 psig) Note 2	10%	≤ 0.56
	40%	≤ 1.40
	60%	≤ 2.03
	100%	≤ 4.20

- (1) The times include 0.2 s delay from de-energizing the scram pilot valves to control rods movement.
- (2) These times are used in pressurization increase events in the DBA category even if the pressure is not greater than 8.75 MPaG (1269 psig) during the rod insertion.

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-5: Defense Lines Inputs Used in Non-LOCA Analyses**

DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-01	Maintain Target Pressure (Performed by RPC)	Steam flow demand is a function of reactor dome pressure	Target dome pressure is normal dome pressure in Table 15.5-3 Design description of RPC is provided in Chapter 7, Section 7.3.3.2, Item 5
DL2-02	Maintain Target Level (Performed by RLC)	RPV level error (function of core power, reactor water level, steam flow, steam flow enthalpy, FW enthalpy)	Target reactor level is NWL (L5) in Table 15.5-3 Design description of RLC is provided in Chapter 7, Section 7.3.3.2, Item 3
DL2-04	Control Rod Block on ATLM	Determination of the thermal limits or soft duty guidelines violation	Design description of ATLM is provided in Chapter 7, Section 7.3.3.2, Item 21
DL2-05	Control Rod Block on MRBM	Indication of potential fuel damage thermal limits being exceeded	Design description of MRBM is provided in Chapter 7, Section 7.3.3.2, Item 22
DL2-08	Anticipatory Hydraulic Scram on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Total time delay = 0.05 s after generator load rejection or turbine trip signal
DL2-09	TBV Fast Open on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Before or at the same time as TCV or TSV closure
DL2-13	Turbine Trip on High Main Condenser Pressure Setpoint 2	High Main Condenser Pressure Setpoint 2	No analytical limit
DL2-14	TBV Closure on High Main Condenser Pressure Setpoint 3	High Main Condenser Pressure Setpoint 3	Set to allow for time for bypass to be open after turbine trip signal for expected rates of condenser pressure increase. Total time delay = 1 s
DL2-19	FW and Condensate Pump Trip on High RPV Level	High RPV Water Level (L8)	L8: 22.277 m (877.047 in.) AVZ Total time delay = 1.0 s
DL2-21	Anticipatory Hydraulic Scram on MSRIV/MSCIV Position	MSRIV/MSCIV Position	<90% open Total time delay (scram) = 0.05 s

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-25	Start Standby FW Pump on Loss of Operating FW Pump	Loss of Operating FW Pump	Analysis allows 10 seconds identify and confirm operating pump is lost and to initiate standby pump. Analysis allows 15 seconds to ramp up the standby pump to full flow
DL2-27	Select Control Rod Run-In on FW Temperature Decrease	FW Temperature Decrease	FW temperature reduction of 16.6°C (30°F) or more. Total time delay = 5 s
DL2-31	ICS Pressure Control on High Reactor Pressure	High Reactor Pressure	Settings are not important for Transient DSA results Design description is provided in Chapter 7, Section 7.3.3.2, Item 6
DL2-39	Anticipatory Hydraulic Scram on High RPV Level	High RPV Level (L8)	See DL2-19
DL3-01	Hydraulic Scram on High RPV Pressure	High RPV Pressure (HP1)	7.787 MPaG (1129.4 psig) Total time delay = 0.7 s
DL3-02	Hydraulic Scram on Low RPV Pressure	Low RPV Pressure (LP1)	5.516 MPaG (800.0 psig) Total time delay = 0.7 s The transient analysis used a lower value this value is required consistent with LOCA analyses.
DL3-03	Hydraulic Scram on Low RPV Level	Low RPV Level (L3)	L3: 19.645 m (773.425 in) AVZ Total time delay = 1.0 s
DL3-04	Hydraulic Scram on High Neutron Flux	High Neutron Flux	125% of rated Total time delay = 0.09 s
DL3-05	Hydraulic Scram on High Simulated Thermal Power	High Simulated Thermal Power	115% of rated Signal Time Constant = 7 s Total time delay = 0.09 s
DL3-11	ICS Train 1 Initiation on High RPV Pressure	High RPV Pressure (HP2)	8.289 MPaG (1202.2 psig) Total time delay = 0.7 s
DL3-12	ICS Train 2 Initiation on High RPV Pressure	High RPV Pressure (HP3)	8.504 MPaG (1233.4 psig) Total time delay = 0.7 s
DL3-13	ICS Train 3 Initiation on High RPV Pressure	High RPV Pressure (HP4)	8.719 MPaG (1264.6 psig) Total time delay = 0.7 s
DL3-14	ICS Initiation on Low RPV Water Level	Low RPV Water Level (L2)	L2: 14.224 m (560.0 in) AVZ Total time delay = 1.0 s



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL3-17	MSRIV/MS CIV Isolation on Low RPV Pressure	Low RPV Pressure (LP1)	5.516 MPaG (800.0 psig) Total time delay = 0.7 s In non-LOCA analysis, this is assumed to occur at a lower pressure that is conservative.
DL3-18	MSRIV/MS CIV Isolation on Low RPV Water Level	Low RPV Water Level (L2)	L2: see DL3-14
DL3-23	FW Isolation on High RPV Water Level	High RPV Water level (L9)	L9: 22.377 m (880.984 in) AVZ Total time delay = 1.0 s
DL4a-12	MSRIV/MS CIV Isolation on Sustained Low FW Flow	Sustained Low FW flow	Analysis allows 70 seconds to confirm sustained low FW flow
DL4a-40	CRD Fast Motor Run-In on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function
DL4a-41	FW Pump/Condensate Pump Trip on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function
DL4a	ICS Tran Initiation on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function

**Table 15.5-6: Sequence of Events for Loss of Feedwater Heating (AOO)**

<b>Time (s)</b>	<b>Event</b>
0	Initiate a 50°C (90°F) temperature reduction in the FW system
5.5	SCRRI inserts control rods on indication of FW temperature reduction
~60	Control rods stop moving
~400	New steady state achieved

**Table 15.5-7: Sequence of Events for Turbine Trip (AOO)**

<b>Time (s)</b>	<b>Events</b>
0.0	Reactor scrams on initiation on turbine trip signal
0.2	Scram rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
>6.0	New steady state

**Table 15.5-8: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (AOO)**

<b>Time (s)</b>	<b>Event</b>
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram on MSRIV position
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
6.0	MSRIV in second steam line is closed
>25.0	High RPV pressure reached, ICS train initiated
>25.0	New Steady state

**Table 15.5-9: Sequence of Events for Loss of Condenser Vacuum (AOO)**

Time (s)	Event
0.0	Anticipatory scram initiated on turbine trip signal caused by loss of condenser vacuum
0.2	Scram rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
24.0	TBVs close on high main condenser pressure
≥24.0	ICS initiation on high RPV pressure
>24.0	New steady state

**Table 15.5-10: Sequence of Events for Loss-of-Preferred Power (AOO)**

Time (s)	Event
0.0	FW pumps lose power
0.0	Anticipatory scram initiated on generator load rejection caused by generator output breakers opening on loss of power
0.2	Scram rods begin to move
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
6.25	TBVs close following loss-of-preferred power
>20.0	ICS initiation on high RPV pressure
>20.0	New steady state

**Table 15.5-11: Sequence of Events for Feedwater Pump Trip (AOO)**

Time (s)	Event
0.0	Initiate FW pump trip
10.0	Standby FW pump starts
25.0	Standby FW pump at 100% rated FW flow
~200	New steady state achieved near 100% power and 100% FW flow

**Table 15.5-12: Sequence of Events for Inadvertent Isolation Condenser Initiation One Train (AOO)**

<b>Time (s)</b>	<b>Event</b>
0	Initiate opening of IC condensate return valve on one train
>150	New steady state achieved near initial conditions

**Table 15.5-13: Sequence of Events for Loss of Feedwater Heating (DBA)**

<b>Time (s)</b>	<b>Event</b>
0	Initiate FW temperature reduction
~90	High STP reached, scram initiated
~110	Low RPV pressure reached, Main Steam isolation initiated
~300	High RPV level L9 reached, FW isolation initiated
> 400	High RPV pressure reached; ICS train initiated (not simulated)

**Table 15.5-14: Sequence of Events for Generator Load Rejection (DBA)**

<b>Time (s)</b>	<b>Event</b>
0.0	TCVs begin to close due to a generator load rejection signal
0.42	High neutron flux reached, Scram initiated
0.62	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
166	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

**Table 15.5-15: Sequence of Loss-of-Preferred Power (DBA)**

Time (s)	Event
0.0	TCV begin to close, and FW pump trips
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
>300	A controlled state is achieved

**Table 15.5-16: RPV Pressure Control Downscale (DBA)**

Time (s)	Event
0.0	TCVs begin to close, and TBVs remain closed
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
155	High RPV pressure is reached, ICS train initiated
167	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

**Table 15.5-17: Closure of All MSRIVs and FW Isolation Valves (DBA)**

Time (s)	Event
0.0	MSRIVs begin to close, and FW isolation valves close
1.09	High Neutron Flux reached, Scram initiated
1.29	Control rods begin to move
122	High RPV pressure reached, ICS initiated
>300	A controlled state is achieved

**Table 15.5-18: Sequence of Events for Feedwater Flow Increase – All Pumps**

<b>Time (s)</b>	<b>Event</b>
0	Initiate instant increase in speed of both FW pumps
~20	High RPV level L9 reached, FW isolation initiated
~30	High STP reached, Scram initiated
~90	Low RPV pressure reached, Main Steam Reactor Pressure Isolation Valve isolation initiated
>200	High RPV pressure reached, ICS train initiated (not simulated)

**Table 15.5-19: Sequence of Events for Condenser Initiation – All Trains**

<b>Time (s)</b>	<b>Events</b>
0	Initiate opening of all IC condensate return valves
~35	High RPV level L9 reached, FW isolation initiated
~55	Low RPV pressure reached, scram, and MSRIV isolation initiated

**Table 15.5-20: Sequence of Events for Loss of Feedwater (DBA)**

<b>Time (s)</b>	<b>Event</b>
0.0	Initiate loss of FW flow
~30	Low RPV level L3 reached, scram initiated
~85	Low RPV pressure reached, MSRIV isolation initiated
~130	Low RPV level L2 reached, all IC Trains initiate
>300	New steady state achieved

**Table 15.5-21: Sequence of Events for RPV Pressure Control Open (DBA)**

Time (s)	Event
0.0	TBVs and TCVs open
~70	Low RPV pressure reached, reactor scram and MSRIV isolation initiated
~225	High RPV level L9 reached, FW isolation initiated
>300	High RPV pressure reached, ICS train initiated (not simulated)

**Table 15.5-22: Not Used**

**Table 15.5-23: Timing of Events for Main Steam Pipe Break Inside Containment, CN-DSA**

Tim (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of main steam pipe break inside the containment	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing	Conservative assumption
1.0	Control rods start to insert on scram initiation	High drywell pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second.
1.0	ICS condensate return valve starts opening	
1.0	CUW stops	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons to diminish. Fission power starts decreasing when the control rods are partially inserted. In addition, voiding in the core due to rapid depressurization also causes reactor power to decrease rapidly.
5.0	RIVs start to close	The delay time is significantly conservative given that containment pressure reaches the isolation setpoint in less than 1 second.
10.0	RIVs are fully closed	
11.0	ICS condensate return valve is fully open	
12.2	Peak containment pressure is reached	
>12.2	Containment pressure starts decreasing	

**Table 15.5-24: Timing of Events for Feedwater Pipe Break  
Inside Containment, CN-DSA**

<b>Time (seconds)</b>	<b>Event</b>	<b>Notes</b>
0.0	Double-ended guillotine rupture of FW pipe break inside containment concurrent with LOPP	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing	Conservative assumption
1.0	Control rods start being inserted on scram initiation	High drywell pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second
1.0	ICS-A and B condensate return valve starts opening	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons
5.0	FWRIVs and CIVs start to close	
10.0	FWRIVs and CIVs are fully closed	
10.1	Peak containment pressure is reached	
11.0	Isolation condenser valves are fully open	



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-25: Timing of Events for Small Steam Pipe Break  
Inside Containment CN-DSA**

<b>Time (seconds)</b>	<b>Event</b>	<b>Notes</b>
0.0	Small steam pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains in initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam line low pressure	
20.6	MSRIVs are fully closed	
63.6	Level decreases to Level 2	
64.6	ICS-A and ICS-B condensate return valves start opening	
74.6	ICS-A and ICS-B condensate return valves are fully open	
234000	Peak containment pressure reached	

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-26: Timing of Events for Small Liquid Pipe Break Inside Containment, CN-DSA**

<b>Time (seconds)</b>	<b>Event</b>	<b>Notes</b>
0.0	Small liquid pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains at initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam pipe low pressure	
20.6	MSRIVs are fully closed	
58.9	Level decreases to Level 2	
59.9	ICS-A and ICS-B condensate return valves start opening	
69.9	ICS-A and ICS-B condensate return valves are fully open	
232800	Peak containment pressure occurs	

**Table 15.5-27: Sequence of Events for Closure of One Main Steam Reactor Pressure Isolation Valve (DEC)**

Time (s)	Event
0.0	Initiate closure of one MSIV
0.8	Anticipatory scram signal on MSRIV position. Scram fails
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
5.8	Feedwater pump trips on high flux after scram signal
5.8	All ICS trains initiate on high flux after scram signal
5.8	CRDM run-in initiation on high flux after scram signal
6.0	MSRIV in second steam line is closed
>300	A controlled state is achieved

**Table 15.5-28: Sequence of Events for Complex Sequence Generator Load Rejection (DEC)**

Time (s)	Event
0.0	Scram initiation on load rejection signal
0.2	Control rods begin to move. Half of the control rods fail to insert
0.25	TCVs begin to close
0.25	TBVs begin to open
>200	A controlled state is achieved

**Table 15.5-29: Sequence of Events for Loss of Condenser Vacuum (DEC)**

Time (s)	Event
0.0	Scram fails on initiation on turbine trip signal caused by loss of condenser vacuum
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
24.0	TBVs close on high main condenser pressure
>300	A controlled state is achieved

**Table 15.5-30: Sequence of Events for Loss of Preferred Power (DEC)**

Time (s)	Event
0.0	FW pumps lose power
0.0	Scram fails on generator load rejection caused by generator output breakers opening on loss of power
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
6.25	TBVs close following loss-of-preferred power
>300	A controlled state is achieved

**Table 15.5-31: Sequence of Events for All Control Rod Withdrawal at Power (ACRW)**

Time (s)	Event
0.1	Initiate withdrawal of all rods
19	High STP reached, scram initiated
63	High RPV Level L9 reached, FW isolation initiated
84	Low RPV pressure reached, Main Steam isolation initiated
> 200	High RPV pressure reached; ICS train initiates on high pressure (not simulated)

**Table 15.5-32: Sequence of Events for Inadvertent Control Rod Withdrawal at Power Single Rod (ICRW)**

Time (s)	Event
0.1	Initiate withdrawal of one rod
>70	Power reaches a stable level
long term	Operators act to control reactor power

**Table 15.5-33: Sequence of Events for Feedwater Isolation (DEC)**

Time (s)	Event
0.0	Initiate FW Isolation
5	FW flow at zero
~30	Low RPV level L3 reached, no scram (DL3 failure)
65	Reactor scram initiated on sustained low FW flow
70	MSRIV isolation initiated on sustained low FW flow
250	ICS Trains initiate on high RPV pressure
>400	Controlled state achieved

**Table 15.5-34: Fuel Handling Accident Sequence of Events**

Sequence of Events	Elapsed Time
The BWRX-300 reactor is shut down for refueling operations that begins 24 hours after shutdown. Over this period the core isotopic inventory decays.	24 hours
During a refueling operation a fuel assembly is moved over the top of the core or fuel pool, and the fuel bundle, grapple, mast, and head fall on top of the core or spent fuel racks.	0
Rods in the dropped bundle and struck bundles fail, releasing the fission gases in the plenum and gap of the damaged rods to the pool water.	0
Fission gases rise through the pool water to refueling operation floor common airspace surrounding the top of the reactor cavity and spent fuel pool.	0
The building ventilation system high radiation alarm alerts all plant personnel to evacuate, and the Reactor Building (RB) air handling system dampers close to isolate the area.	0
Plant workers evacuate the RB and adjacent areas.	15 minutes

**Table 15.5-35: BWRX-300 Core Parameters**

Parameter/Description	Value
Thermal Power	870(MWth)
Core Size (Number of Bundles)	240
Fuel Type	GNF2
Bundle Average Enrichment	3.84-4.68 <sup>w</sup> ‰ U235
Core Average Exposure	38,000 MWd/Mt

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-36: BWRX-300 Core Inventory 24 Hours After Shutdown**

Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)	Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)
Co-58	3.08E+02	1.14E+07	Te-131m	2.31E+03	8.55E+07
Co-60	6.51E+02	2.41E+07	Te-132	3.13E+04	1.16E+09
Kr-85	5.40E+02	2.00E+07	I-131	2.54E+04	9.40E+08
Kr-85m	1.78E+02	6.59E+06	I-132	3.23E+04	1.20E+09
Kr-87	2.91E-02	1.08E+03	I-133	2.54E+04	9.40E+08
Kr-88	5.56E+01	2.06E+06	I-134	1.36E-03	5.03E+01
Rb-86	7.40E+01	2.74E+06	I-135	4.16E+03	1.54E+08
Sr-89	2.56E+04	9.47E+08	Xe-133	5.16E+04	1.91E+09
Sr-90	4.48E+03	1.66E+08	Xe-135	1.49E+04	5.51E+08
Sr-91	5.70E+03	2.11E+08	Cs-134	9.46E+03	3.50E+08
Sr-92	7.64E+01	2.83E+06	Cs-136	2.60E+03	9.62E+07
Y-90	4.62E+03	1.71E+08	Cs-137	5.97E+03	2.21E+08
Y-91	3.32E+04	1.23E+09	Ba-139	3.20E-01	1.18E+04
Y-92	1.13E+03	4.18E+07	Ba-140	4.51E+04	1.67E+09
Y-93	8.00E+03	2.96E+08	La-140	4.88E+04	1.81E+09
Zr-95	4.73E+04	1.75E+09	La-141	7.08E+02	2.62E+07
Zr-97	1.84E+04	6.81E+08	La-142	1.03E+00	3.81E+04
Nb-95	4.80E+04	1.78E+09	Ce-141	4.44E+04	1.64E+09
Mo-99	4.00E+04	1.48E+09	Ce-143	2.55E+04	9.44E+08
Tc-99m	3.81E+04	1.41E+09	Ce-144	3.82E+04	1.41E+09
Ru-103	4.33E+04	1.60E+09	Pr-143	4.02E+04	1.49E+09
Ru-105	7.65E+02	2.83E+07	Nd-147	1.70E+04	6.29E+08
Ru-106	2.05E+04	7.59E+08	Np-239	4.53E+05	1.68E+10
Rh-105	2.15E+04	7.96E+08	Pu-238	2.28E+02	8.44E+06
Sb-127	2.63E+03	9.73E+07	Pu-239	1.87E+01	6.92E+05
Sb-129	1.95E+02	7.22E+06	Pu-240	2.70E+01	9.99E+05
Te-127	2.87E+03	1.06E+08	Pu-241	7.83E+03	2.90E+08
Te-127m	4.19E+02	1.55E+07	Am-241	1.47E+01	5.44E+05
Te-129	1.08E+03	4.00E+07	Cm-242	3.03E+03	1.12E+08
Te-129m	1.31E+03	4.85E+07	Cm-244	2.15E+02	7.96E+06

**Table 15.5-37: Non-LOCA Fraction of Fission Product Inventory in Gap**

Group	Gap Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

**Table 15.5-38: BWRX-300 FHA Activity Released from Fuel**

Nuclide	Core Inventory (Ci/MWth)	Core Inventory (MBq/MWth)	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)
Kr-85m	1.78E+02	6.59E+06	8.20E+01	3.03E+06
Kr-85	5.40E+02	2.00E+07	4.98E+02	1.84E+07
Kr-87	2.91E-02	1.08E+03	1.34E-02	4.96E+02
Kr-88	5.56E+01	2.06E+06	2.56E+01	9.48E+05
I-131	2.54E+04	9.40E+08	1.87E+04	6.93E+08
I-132	3.23E+04	1.20E+09	1.49E+04	5.51E+08
I-133	2.54E+04	9.40E+08	1.17E+04	4.33E+08
I-134	1.36E-03	5.03E+01	6.27E-04	2.32E+01
I-135	4.16E+03	1.54E+08	1.92E+03	7.09E+07
Xe-133	5.16E+04	1.91E+09	2.38E+04	8.80E+08
Xe-135	1.49E+04	5.51E+08	6.87E+03	2.54E+08



**Table 15.5-39: BWRX-300 Fuel Handling Accident (FHA) Activity Released from the Reactor Cavity Pool**

Nuclide	Activity Released to the Environment (TBq)
Kr-85m	2.80E+00
Kr-85	1.80E+01
Kr-87	3.70E-04
Kr-88	8.30E-01
I-131	3.40E+00
I-132	2.30E+00
I-133	2.10E+00
I-134	7.90E-08
I-135	3.30E-01
Xe-133	8.80E+02
Xe-135	2.40E+02

**Table 15.5-40: Not Used**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-41: FWLB Line Break Accident Airborne Release Source Term**

<b>Nuclide</b>	<b>Equilibrium Release (TBq)</b>	<b>Spike Release (TBq)</b>
Kr-83m	1.76E-03	1.76E-03
Kr-85m	4.84E-05	4.84E-05
Kr-85	1.17E-05	1.17E-05
Kr-87	3.30E-04	3.30E-04
Kr-88	1.83E-04	1.83E-04
Kr-89	6.30E-02	6.30E-02
Xe-131m	1.03E-05	1.03E-05
Xe-133m	6.01E-06	6.01E-06
Xe-133	8.80E-05	8.80E-05
Xe-135m	1.61E-03	1.61E-03
Xe-135	8.80E-04	8.80E-04
Xe-137	3.01E-03	3.01E-03
Xe-138	8.06E-03	8.06E-03
I-131	4.89E-02	9.60E-01
I-132	5.25E-01	1.03E+01
I-133	3.62E-01	7.24E+00
I-134	1.58E+00	3.08E+01
I-135	7.24E-01	1.43E+01
Rb-89	8.50E-01	8.50E-01
Cs-134	1.03E-02	1.03E-02
Cs-136	8.14E-03	8.14E-03
Cs-137	1.57E-02	1.57E-02
Cs-138	9.05E-01	9.05E-01
Ba-137m	1.57E-02	1.57E-02
HTO	9.79E-03	9.79E-03
Cr-51	5.43E-04	5.43E-04
Mn-54	2.71E-04	2.71E-04
Fe-59	1.45E-04	1.45E-04
Co-60	2.53E-04	2.53E-04
Cu-64	2.71E-03	2.71E-03
Sr-89	1.14E-03	1.14E-03
Sr-90	5.79E-05	5.79E-05

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Y-90	5.79E-05	5.79E-05
Sr-91	7.42E-01	7.42E-01
Sr-92	1.70E+00	1.70E+00
Y-91	1.52E-02	1.52E-02
Y-92	5.07E-01	5.07E-01
Y-93	5.25E-02	5.25E-02
Zr-95	3.08E-02	3.08E-02
Nb-95	3.08E-02	3.08E-02
Mo-99	1.50E-01	1.50E-01
Tc-99m	1.50E-01	1.50E-01
Ru-103	7.60E-03	7.60E-03
Rh-103m	7.60E-03	7.60E-03
Ru-106	1.14E-03	1.14E-03
Rh-106	1.14E-03	1.14E-03
Te-129m	1.52E-02	1.52E-02
Te-131m	1.99E-02	1.99E-02
Te-132	3.62E-03	3.62E-03
Ba-140	1.59E-01	1.59E-01
La-140	1.59E-01	1.59E-01
Ce-141	7.60E-03	7.60E-03
Ce-144	1.14E-03	1.14E-03
Pr-144	1.14E-03	1.14E-03
Np-239	1.23E-01	1.23E-01

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-42: ICS Line Break Accident Airborne Release Source Term**

<b>Nuclide</b>	<b>Equilibrium Release (TBq)</b>	<b>Spike Release (TBq)</b>
Kr-83m	9.62E-04	9.62E-04
Kr-85m	2.65E-05	2.65E-05
Kr-85	6.41E-06	6.41E-06
Kr-87	1.80E-04	1.80E-04
Kr-88	1.00E-04	1.00E-04
Kr-89	3.45E-02	3.45E-02
Xe-131m	5.61E-06	5.61E-06
Xe-133m	3.29E-06	3.29E-06
Xe-133	4.81E-05	4.81E-05
Xe-135m	8.82E-04	8.82E-04
Xe-135	4.81E-04	4.81E-04
Xe-137	1.64E-03	1.64E-03
Xe-138	4.41E-03	4.41E-03
I-131	2.20E-05	4.41E-04
I-132	2.20E-04	4.41E-03
I-133	1.60E-04	3.21E-03
I-134	6.81E-04	1.40E-02
I-135	3.05E-04	6.01E-03
Rb-89	1.88E-05	1.88E-05
Cs-134	2.28E-07	2.28E-07
Cs-136	1.80E-07	1.80E-07
Cs-137	3.49E-07	3.49E-07
Cs-138	2.00E-05	2.00E-05
Ba-137m	3.49E-07	3.49E-07
HTO	2.08E-04	2.08E-04
Cr-51	1.20E-08	1.20E-08
Mn-54	6.01E-09	6.01E-09
Fe-59	3.21E-09	3.21E-09
Co-60	5.61E-09	5.61E-09
Cu-64	6.01E-08	6.01E-08
Sr-89	2.53E-08	2.53E-08
Sr-90	1.28E-09	1.28E-09

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Y-90	1.28E-09	1.28E-09
Sr-91	1.64E-05	1.64E-05
Sr-92	3.77E-05	3.77E-05
Y-91	3.37E-07	3.37E-07
Y-92	1.12E-05	1.12E-05
Y-93	1.16E-06	1.16E-06
Zr-95	6.81E-07	6.81E-07
Nb-95	6.81E-07	6.81E-07
Mo-99	3.33E-06	3.33E-06
Tc-99m	3.33E-06	3.33E-06
Ru-103	1.68E-07	1.68E-07
Rh-103m	1.68E-07	1.68E-07
Ru-106	2.53E-08	2.53E-08
Rh-106	2.53E-08	2.53E-08
Te-129m	3.37E-07	3.37E-07
Te-131m	4.41E-07	4.41E-07
Te-132	8.02E-08	8.02E-08
Ba-140	3.53E-06	3.53E-06
La-140	3.53E-06	3.53E-06
Ce-141	1.68E-07	1.68E-07
Ce-144	2.53E-08	2.53E-08
Pr-144	2.53E-08	2.53E-08
Np-239	2.73E-06	2.73E-06

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-43A: Instrument Line Break (ILB) Accident Airborne Equilibrium Release Source Term**

<b>Radionuclides</b>	<b>Activity Released to the Environment in 0-1800 seconds(TBq)</b>	<b>Activity Released to the Environment in 1800 - 9700 seconds(TBq)</b>	<b>Activity Released to the Environment in 9700 - 15500 seconds(TBq)</b>	<b>Activity Released to the Environment in 15500 - 20500 seconds(TBq)</b>
Kr-83m	2.12E-03	5.56E-03	1.35E-03	5.82E-04
Kr-85m	5.82E-05	1.53E-04	3.72E-05	1.60E-05
Kr-85	1.41E-05	3.71E-05	9.01E-06	3.88E-06
Kr-87	3.97E-04	1.04E-03	2.53E-04	1.09E-04
Kr-88	2.20E-04	5.80E-04	1.41E-04	6.07E-05
Kr-89	7.58E-02	1.99E-01	4.84E-02	2.09E-02
Xe-131m	1.23E-05	3.25E-05	7.88E-06	3.40E-06
Xe-133m	7.23E-06	1.90E-05	4.62E-06	1.99E-06
Xe-133	1.06E-04	2.78E-04	6.76E-05	2.91E-05
Xe-135m	1.94E-03	5.10E-03	1.24E-03	5.34E-04
Xe-135	1.06E-03	2.78E-03	6.76E-04	2.91E-04
Xe-137	3.62E-03	9.51E-03	2.31E-03	9.95E-04
Xe-138	9.70E-03	2.55E-02	6.19E-03	2.67E-03
I-131	6.29E-03	2.03E-02	9.49E-03	4.09E-03
I-132	6.76E-02	2.18E-01	1.02E-01	4.39E-02
I-133	4.66E-02	1.50E-01	7.03E-02	3.03E-02
I-134	2.03E-01	6.53E-01	3.06E-01	1.32E-01
I-135	9.32E-02	3.00E-01	1.41E-01	6.06E-02
Rb-89	1.09E-01	3.51E-01	1.65E-01	7.10E-02
Cs-134	1.32E-03	4.25E-03	2.00E-03	8.61E-04

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Radionuclides</b>	<b>Activity Released to the Environment in 0- 1800 seconds(TBq)</b>	<b>Activity Released to the Environment in 1800 - 9700 seconds(TBq)</b>	<b>Activity Released to the Environment in 9700 - 15500 seconds(TBq)</b>	<b>Activity Released to the Environment in 15500 - 20500 seconds(TBq)</b>
Cs-136	1.04E-03	3.36E-03	1.58E-03	6.79E-04
Cs-137	2.01E-03	6.49E-03	3.05E-03	1.31E-03
Cs-138	1.16E-01	3.73E-01	1.75E-01	7.55E-02
Ba-137m	2.01E-03	6.49E-03	3.05E-03	1.31E-03
HTO	1.66E-03	5.08E-03	2.11E-03	9.11E-04
Cr-51	6.94E-05	2.24E-04	1.05E-04	4.53E-05
Mn-54	3.47E-05	1.12E-04	5.25E-05	2.26E-05
Fe-59	1.85E-05	5.97E-05	2.80E-05	1.21E-05
Co-60	3.24E-05	1.04E-04	4.90E-05	2.11E-05
Cu-64	3.47E-04	1.12E-03	5.25E-04	2.26E-04
Sr-89	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Sr-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Y-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Sr-91	9.49E-02	3.06E-01	1.44E-01	6.19E-02
Sr-92	2.17E-01	7.01E-01	3.29E-01	1.42E-01
Y-91	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Y-92	6.48E-02	2.09E-01	9.81E-02	4.23E-02
Y-93	6.71E-03	2.16E-02	1.02E-02	4.38E-03
Zr-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Nb-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Mo-99	1.92E-02	6.19E-02	2.91E-02	1.25E-02

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Radionuclides</b>	<b>Activity Released to the Environment in 0- 1800 seconds(TBq)</b>	<b>Activity Released to the Environment in 1800 - 9700 seconds(TBq)</b>	<b>Activity Released to the Environment in 9700 - 15500 seconds(TBq)</b>	<b>Activity Released to the Environment in 15500 - 20500 seconds(TBq)</b>
Tc-99m	1.92E-02	6.19E-02	2.91E-02	1.25E-02
Ru-103	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Rh-103m	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ru-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Rh-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Te-129m	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Te-131m	2.54E-03	8.21E-03	3.85E-03	1.66E-03
Te-132	4.63E-04	1.49E-03	7.01E-04	3.02E-04
Ba-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
La-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
Ce-141	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ce-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Pr-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Np-239	1.57E-02	5.07E-02	2.38E-02	1.03E-02



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-43B: Instrument Line Break (ILB) Accident Airborne Iodine Spike Release Source Term**

<b>Radionuclides</b>	<b>Activity Released to the Environment in 0-1800 seconds (TBq)</b>	<b>Activity Released to the Environment in 1800 - 9700 seconds (TBq)</b>	<b>Activity Released to the Environment in 9700 - 15500 seconds (TBq)</b>	<b>Activity Released to the Environment in 15500 - 20500 seconds (TBq)</b>
Kr-83m	2.12E-03	5.56E-03	1.35E-03	5.82E-04
Kr-85m	5.82E-05	1.53E-04	3.72E-05	1.60E-05
Kr-85	1.41E-05	3.71E-05	9.01E-06	3.88E-06
Kr-87	3.97E-04	1.04E-03	2.53E-04	1.09E-04
Kr-88	2.20E-04	5.80E-04	1.41E-04	6.07E-05
Kr-89	7.58E-02	1.99E-01	4.84E-02	2.09E-02
Xe-131m	1.23E-05	3.25E-05	7.88E-06	3.40E-06
Xe-133m	7.23E-06	1.90E-05	4.62E-06	1.99E-06
Xe-133	1.06E-04	2.78E-04	6.76E-05	2.91E-05
Xe-135m	1.94E-03	5.10E-03	1.24E-03	5.34E-04
Xe-135	1.06E-03	2.78E-03	6.76E-04	2.91E-04
Xe-137	3.62E-03	9.51E-03	2.31E-03	9.95E-04
Xe-138	9.70E-03	2.55E-02	6.19E-03	2.67E-03
I-131	1.24E-01	3.98E-01	1.86E-01	8.03E-02
I-132	1.33E+00	4.28E+00	2.00E+00	8.63E-01
I-133	9.32E-01	3.00E+00	1.41E+00	6.06E-01
I-134	3.96E+00	1.28E+01	5.97E+00	2.57E+00
I-135	1.84E+00	5.93E+00	2.78E+00	1.20E+00
Rb-89	1.09E-01	3.51E-01	1.65E-01	7.10E-02
Cs-134	1.32E-03	4.25E-03	2.00E-03	8.61E-04

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Radionuclides	Activity Released to the Environment in 0-1800 seconds (TBq)	Activity Released to the Environment in 1800 - 9700 seconds (TBq)	Activity Released to the Environment in 9700 - 15500 seconds (TBq)	Activity Released to the Environment in 15500 - 20500 seconds (TBq)
Cs-136	1.04E-03	3.36E-03	1.58E-03	6.79E-04
Cs-137	2.01E-03	6.49E-03	3.05E-03	1.31E-03
Cs-138	1.16E-01	3.73E-01	1.75E-01	7.55E-02
Ba-137m	2.01E-03	6.49E-03	3.05E-03	1.31E-03
HTO	1.66E-03	5.08E-03	2.11E-03	9.11E-04
Cr-51	6.94E-05	2.24E-04	1.05E-04	4.53E-05
Mn-54	3.47E-05	1.12E-04	5.25E-05	2.26E-05
Fe-59	1.85E-05	5.97E-05	2.80E-05	1.21E-05
Co-60	3.24E-05	1.04E-04	4.90E-05	2.11E-05
Cu-64	3.47E-04	1.12E-03	5.25E-04	2.26E-04
Sr-89	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Sr-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Y-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Sr-91	9.49E-02	3.06E-01	1.44E-01	6.19E-02
Sr-92	2.17E-01	7.01E-01	3.29E-01	1.42E-01
Y-91	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Y-92	6.48E-02	2.09E-01	9.81E-02	4.23E-02
Y-93	6.71E-03	2.16E-02	1.02E-02	4.38E-03
Zr-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Nb-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Mo-99	1.92E-02	6.19E-02	2.91E-02	1.25E-02

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

<b>Radionuclides</b>	<b>Activity Released to the Environment in 0- 1800 seconds (TBq)</b>	<b>Activity Released to the Environment in 1800 - 9700 seconds (TBq)</b>	<b>Activity Released to the Environment in 9700 - 15500 seconds (TBq)</b>	<b>Activity Released to the Environment in 15500 - 20500 seconds (TBq)</b>
Tc-99m	1.92E-02	6.19E-02	2.91E-02	1.25E-02
Ru-103	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Rh-103m	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ru-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Rh-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Te-129m	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Te-131m	2.54E-03	8.21E-03	3.85E-03	1.66E-03
Te-132	4.63E-04	1.49E-03	7.01E-04	3.02E-04
Ba-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
La-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
Ce-141	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ce-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Pr-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Np-239	1.57E-02	5.07E-02	2.38E-02	1.03E-02

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-44: Main Steam Line Break Accident Airborne Release Source Term**

Isotope	Equilibrium Iodine	Pre-Incident Iodine Spike
	TBq	TBq
Kr-83m	5.76E-03	5.76E-03
Kr-85m	1.58E-04	1.58E-04
Kr-85	3.84E-05	3.84E-05
Kr-87	1.08E-03	1.08E-03
Kr-88	6.00E-04	6.00E-04
Kr-89	2.06E-01	2.06E-01
Xe-131m	3.36E-05	3.36E-05
Xe-133m	1.97E-05	1.97E-05
Xe-133	2.88E-04	2.88E-04
Xe-135m	5.28E-03	5.28E-03
Xe-135	2.88E-03	2.88E-03
Xe-137	9.84E-03	9.84E-03
Xe-138	2.64E-02	2.64E-02
I-131	4.30E-02	8.44E-01
I-132	4.62E-01	9.08E+00
I-133	3.19E-01	6.37E+00
I-134	1.39E+00	2.71E+01
I-135	6.37E-01	1.26E+01
Rb-89	7.46E-01	7.46E-01
Cs-134	9.05E-03	9.05E-03
Cs-136	7.15E-03	7.15E-03
Cs-137	1.38E-02	1.38E-02
Cs-138	7.94E-01	7.94E-01
Ba-137m	1.38E-02	1.38E-02
HTO	9.50E-03	9.50E-03
Cr-51	4.76E-04	4.76E-04
Mn-54	2.38E-04	2.38E-04
Fe-59	1.27E-04	1.27E-04
Co-60	2.22E-04	2.22E-04
Cu-64	2.38E-03	2.38E-03
Sr-89	1.00E-03	1.00E-03

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

Isotope	Equilibrium Iodine	Pre-Incident Iodine Spike
	TBq	TBq
Sr-90	5.08E-05	5.08E-05
Y-90	5.08E-05	5.08E-05
Sr-91	6.51E-01	6.51E-01
Sr-92	1.49E+00	1.49E+00
Y-91	1.33E-02	1.33E-02
Y-92	4.45E-01	4.45E-01
Y-93	4.61E-02	4.61E-02
Zr-95	2.70E-02	2.70E-02
Nb-95	2.70E-02	2.70E-02
Mo-99	1.32E-01	1.32E-01
Tc-99m	1.32E-01	1.32E-01
Ru-103	6.67E-03	6.67E-03
Rh-103m	6.67E-03	6.67E-03
Ru-106	1.00E-03	1.00E-03
Rh-106	1.00E-03	1.00E-03
Te-129m	1.33E-02	1.33E-02
Te-131m	1.75E-02	1.75E-02
Te-132	3.18E-03	3.18E-03
Ba-140	1.40E-01	1.40E-01
La-140	1.40E-01	1.40E-01
Ce-141	6.67E-03	6.67E-03
Ce-144	1.00E-03	1.00E-03
Pr-144	1.00E-03	1.00E-03
Np-239	1.08E-01	1.08E-01

**Table 15.5-45: Description of Criticality Accident Parameters**

Details	Parameters
Site/Date	SACLAY (1960/03/15)
Critical Mass	2200 kg
Critical Geometry	Oxide Fuel in Water
Total # Fissions	$3 \times 10^{18}$
Fission Excursion	< 180 sec.

**Table 15.5-46: Out of Core Criticality Scenario Results**

Total Number of Fissions	$4 \times 10^{19}$ fissions
Excursion Duration	< 180 s
Total Radiation Dose (350 m)	0.4 mSv

**Table 15.5-47: Protective Actions and Generic Criteria for OOC**

Protective Actions	Generic Criteria
Evacuation	50 mSv effective dose in 7 days
Sheltering	5 mSv effective dose in 1 day
Temporary Relocation	50 mSv in the first year

**Table 15.5-48: Conservatisms Used in the Non-LOCA DSA**

Plant State	Conservatism	
	Code	Plant Parameters and System Performances
AOO Base Line Analyses	Best Estimate	<ul style="list-style-type: none"> <li>Rated power initial conditions</li> <li>Conservative setpoints and plant performance parameters</li> <li>Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method</li> <li>DL2 functions are credited in the analyses, but DL3 functions are credited if needed to meet acceptance criteria</li> </ul>
AOO Conservative	Best Estimate	Fault evaluations indicate that failures in DL2 result in DBA events
DBA Baseline	Best Estimate	<ul style="list-style-type: none"> <li>Rated power initial conditions</li> <li>Conservative setpoints and plant performance parameters</li> <li>Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method</li> <li>DL2 and DL3 functions are credited if needed to meet acceptance criteria</li> </ul>
DBA Conservative	Graded Approach	<ul style="list-style-type: none"> <li>Conservative initial conditions are established (e.g., bias fuel and reactor pressure initial) for events that are limiting</li> <li>Conservative setpoints and plant performance parameters</li> <li>Analysis method uncertainties are addressed with a graded approach depending on margin to the derived acceptance criteria</li> </ul>
DEC	Best Estimate	<ul style="list-style-type: none"> <li>Rated power initial conditions are used</li> <li>Nominal setpoints and plant performance parameters are used (conservative setpoints and plant performance may be used for convenience/simplification)</li> <li>Sensitivity analyses performed to understand cliff edge effects</li> </ul>

**Table 15.5-49: DL3 Functions Credited in Conservative LOCA Analyses**

<b>Credited Function</b>	<b>Action</b>	<b>Signal</b>	<b>Analytical Setpoint</b>
DL3-02	Hydraulic scram	Low steam pipe pressure	5.617 MPa
DL3-07	Hydraulic scram	High containment pressure	18.5 kPaG (2.8 psig)
DL3-09	Hydraulic scram	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm (0.75 in) in diameter (Note 1)
DL3-14	ICS initiation	Low RPV water level	14.22 m
DL3-15	ICS initiation	High Containment pressure	18.5 kPaG (2.8 psig)
DL3-16	ICS initiation	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm (0.75 in) in diameter (Note 1)
DL3-17	MS RIV closure (Note 2)	Low steam pipe pressure	5.617 MPa
DL3-18	MS RIV closure (Note 2)	Low RPV water level	14.22 m
DL3-22	Reactor and Containment Isolation Valve closure (Note 2)	High Containment pressure	18.5 kPaG (2.8 psig) (Note 1)
DL3-20 DL3-21	MS RIV closure	Line break indication in MSL, FW or SDC	Within 1 seconds for breaks larger than 19 mm (0.75 in) in diameter
DL3-25	FW and SDC RIV closure	Line break indication in FW or SDC	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter
DL3-26	CUW RIV closure	Line break indication in CUW	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter
DL3-27 DL3-28 DL3-29	ICS RIV closure of the broken ICS train	Line break indication in the respective ICS trains	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter

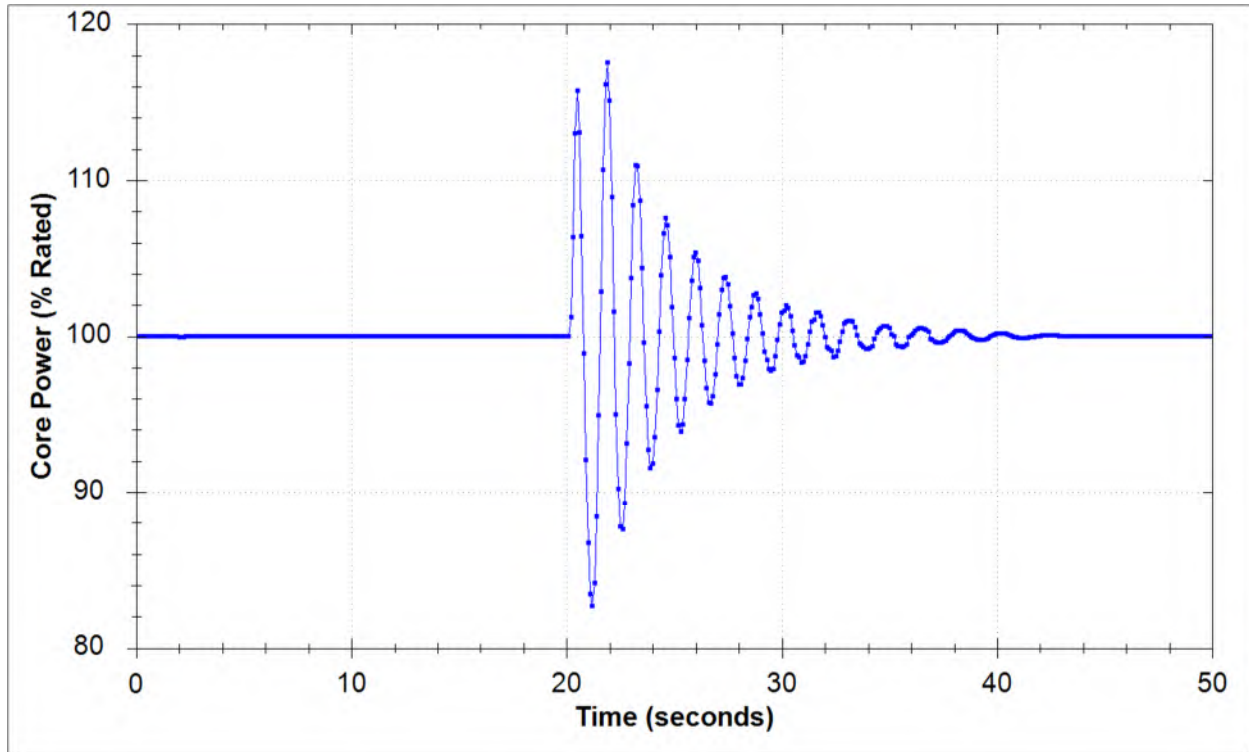
- (1) This setpoint is reached in less than 1 s in large break cases.
- (2) For large breaks, isolation valves are assumed to start closing with a 5 second delay from the time of pipe break and are fully closed in 10 seconds. For small breaks, the isolation valves start closing with a 5 second delay after the setpoint is reached and are fully closed in another 5 seconds. Containment isolation is credited only for FW pipe breaks. Isolation functions may also include CIVs. This table shows only the valves credited to close by the isolation signal.



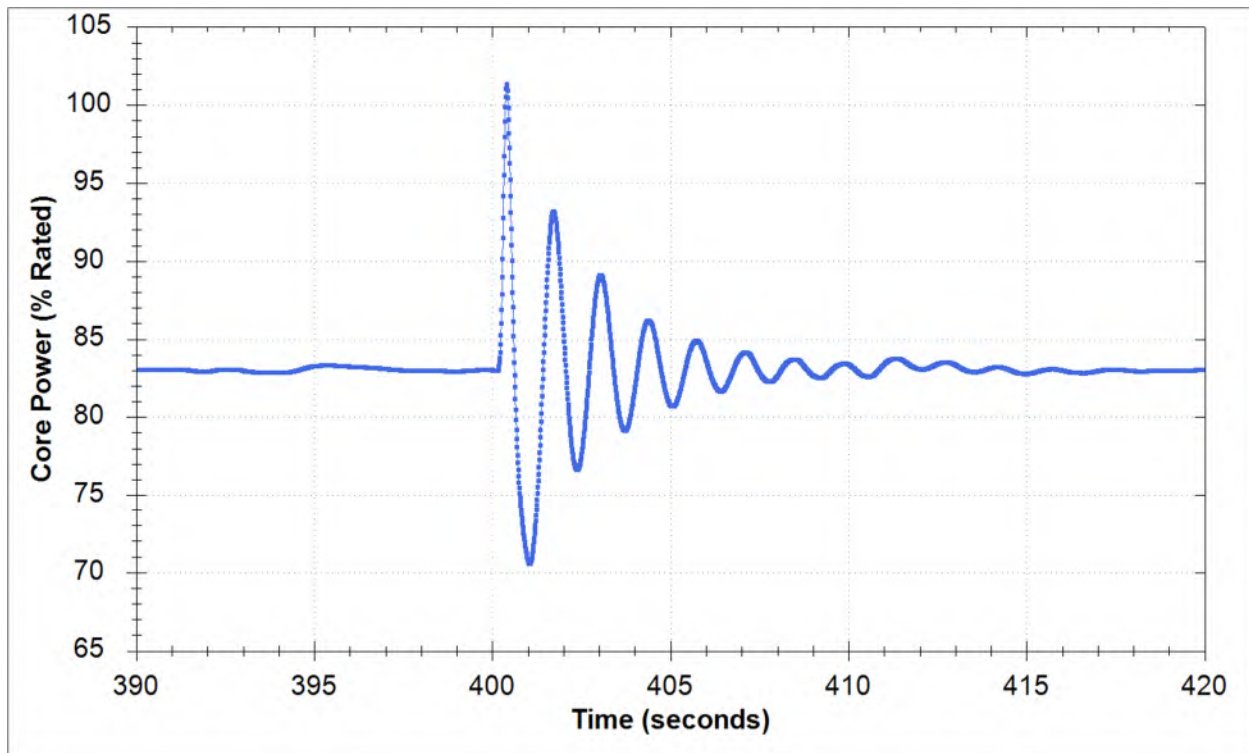
NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 15.5-50: DL2 and DL4a Functions Credited in DEC LOCA Analyses**

<b>DL3 Function Failure</b>	<b>Action</b>	<b>DL2 or DL4a Credited Function</b>	<b>Notes</b>
DL3-02	Hydraulic scram on low steam pipe pressure	DL2-08 on turbine trip demand	DL2-08 is initiated on LOPP, and a faster scram than DL3-02 credited in CN-DSA.
DL3-07	Hydraulic scram on high containment pressure	DL4a-05	
DL3-09	Hydraulic scram on Line Break Indication (MS, FW, ICS)	DL4a-26	
DL3-14	ICS initiation on Low RPV water level	DL4a-33	
DL3-15	ICS initiation on High Containment pressure	DL4a-11	
DL3-16	ICS initiation on Line Break Indication (MS, FW, ICS)	DL4a-27 and DL4a-28	
DL3-17	MS RIV closure on Low steam pipe pressure	DL2-41	DL3-17 is credited in small break cases for limiting inventory losses to the turbine. In DL2, the pressure controller throttles the flow until DL2-41 closes the MS RIVs. This combination of functions results in less inventory losses than the CN-DSA sequences.
DL3-18	MS RIV closure on Low RPV water level	DL4a-34	
DL3-22	Reactor and Containment Isolation Valve closure on High Containment pressure	DL4a-16	
DL3-20 DL3-21	MSRIV closure on Line break indication in MSL, FW or SDC	DL4a-14 and DL4a-15	
DL3-25	FW and SDC RIV closure on line break indication in FW or SDC	DL4a-17	
DL3-27 DL3-28 DL3-29	ICRIV closure of the broken ICS train line break indication in the respective ICS train	None	See discussion in 15.2.4.8.1.



**Figure 15.5-1: MOC Transient Results for Core-Wide Stability Analysis**



**Figure 15.5-2: MOC Transient Results for LFWH with SCCRI Core-Wide Stability Analysis**

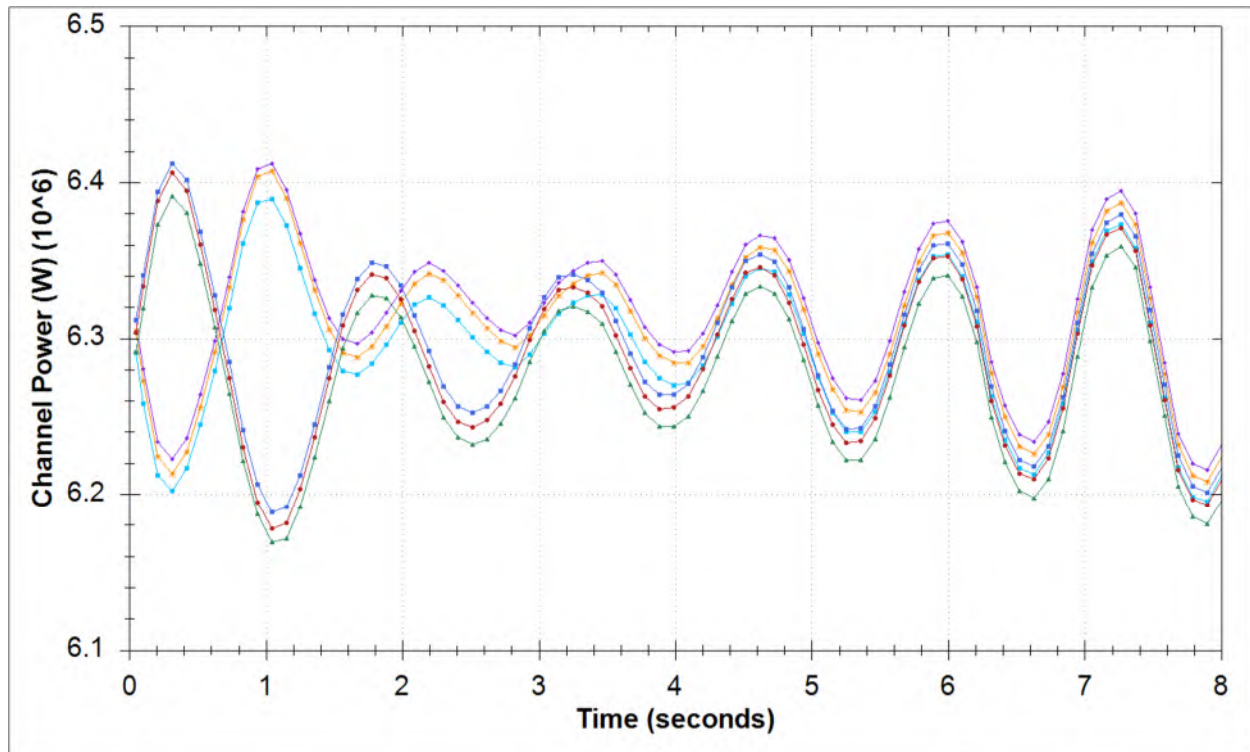


Figure 15.5-3: Regional Mode Stability Response at MOC for FW Temperature of 241.9°C

Figure 15.5-4: Not Used

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

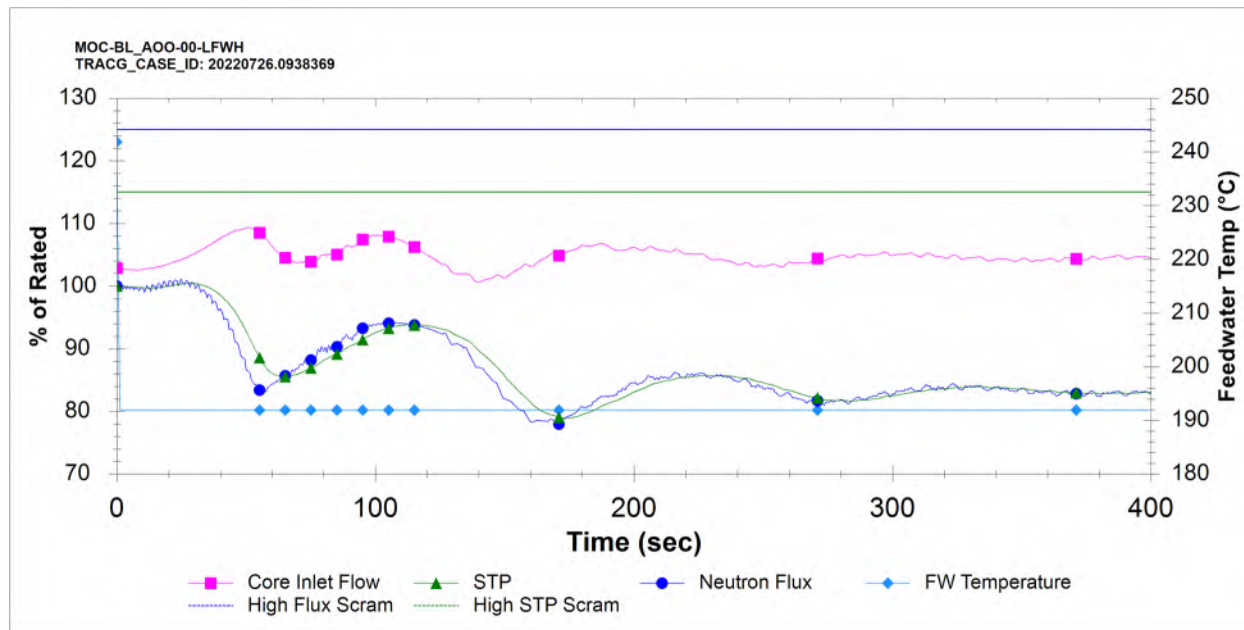
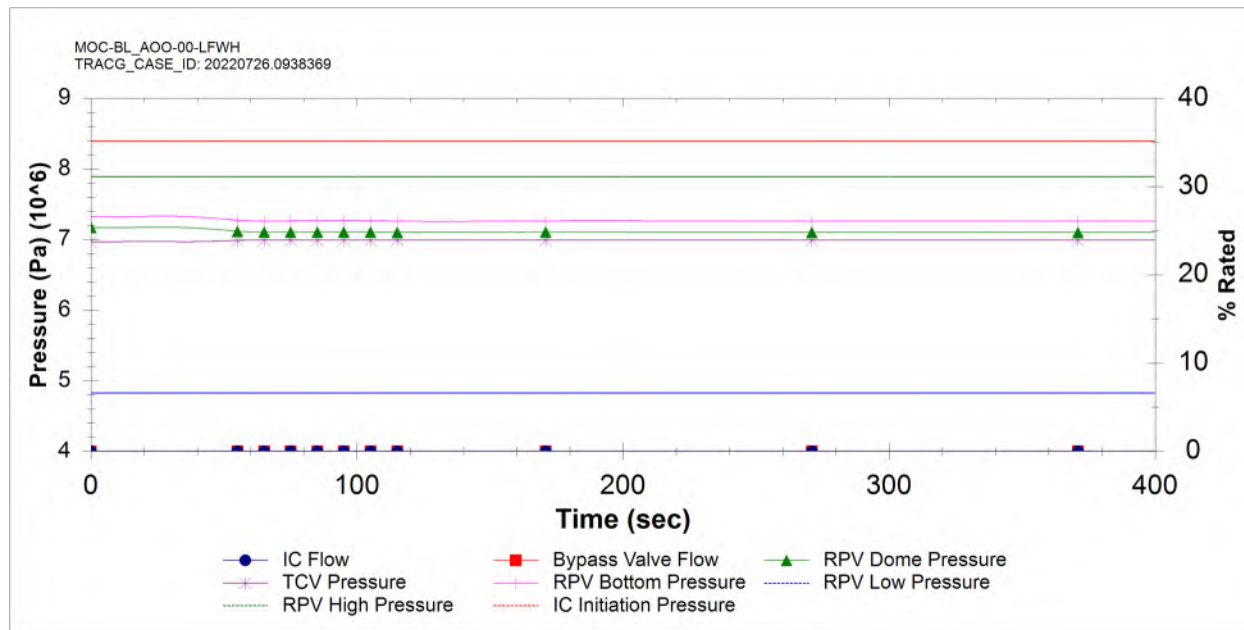


Figure 15.5-5: Loss of Feedwater Heating (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-6: Loss of Feedwater Heating (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

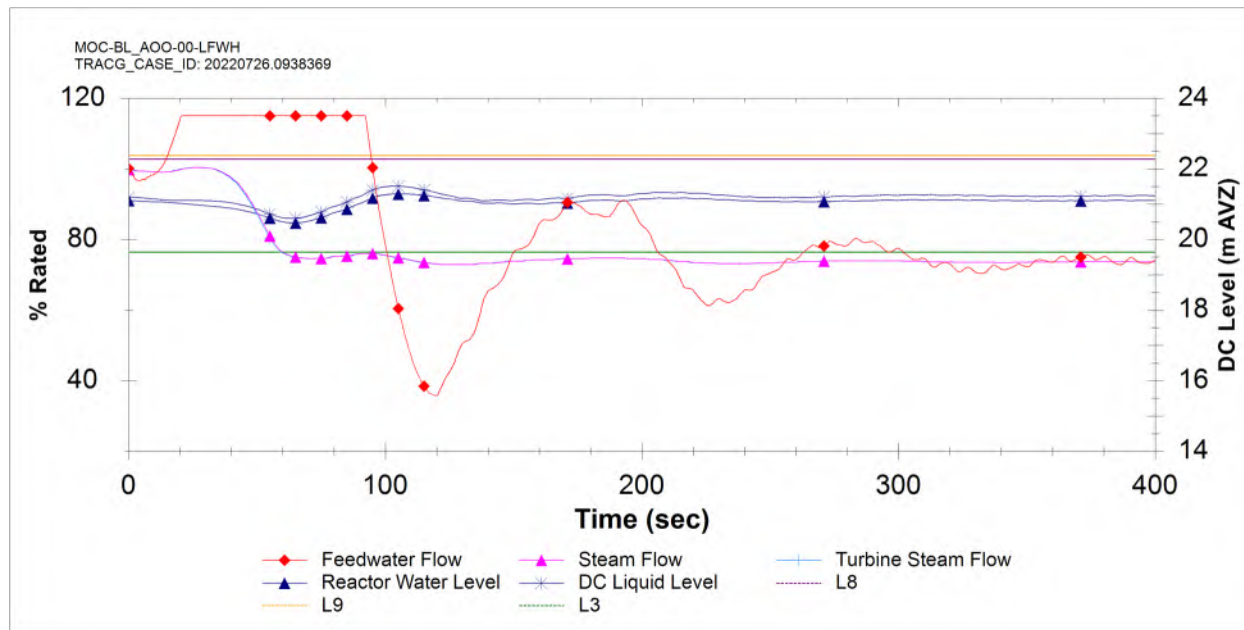


Figure 15.5-7: Loss of Feedwater Heating (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

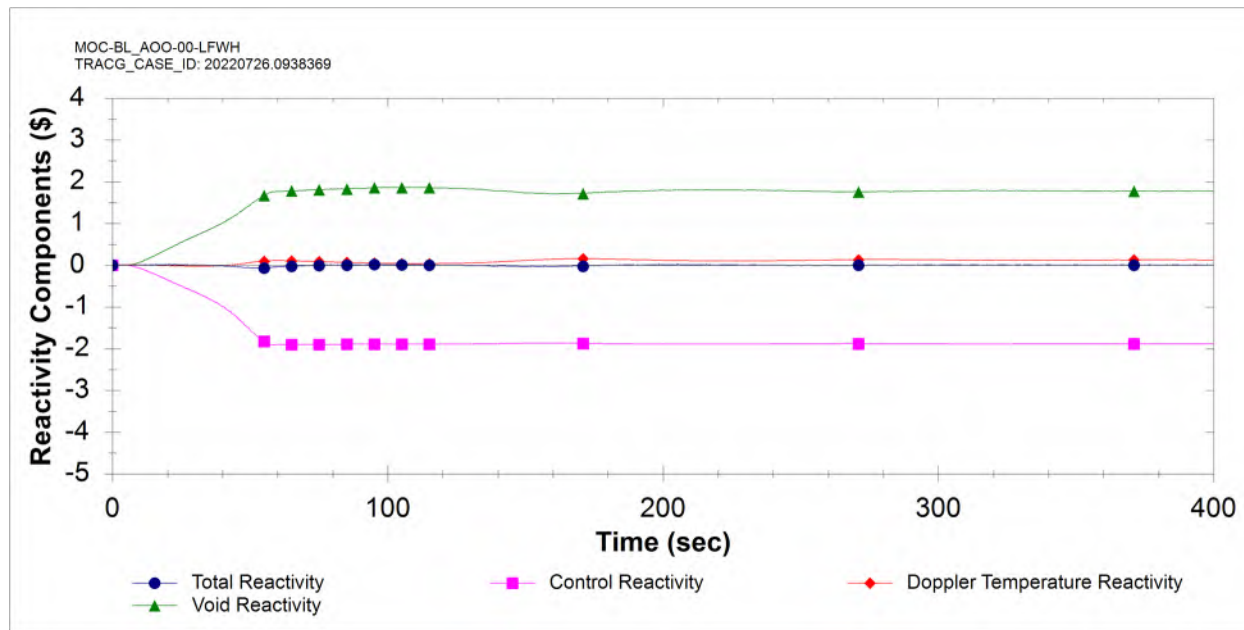


Figure 15.5-8: Loss of Feedwater Heating (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

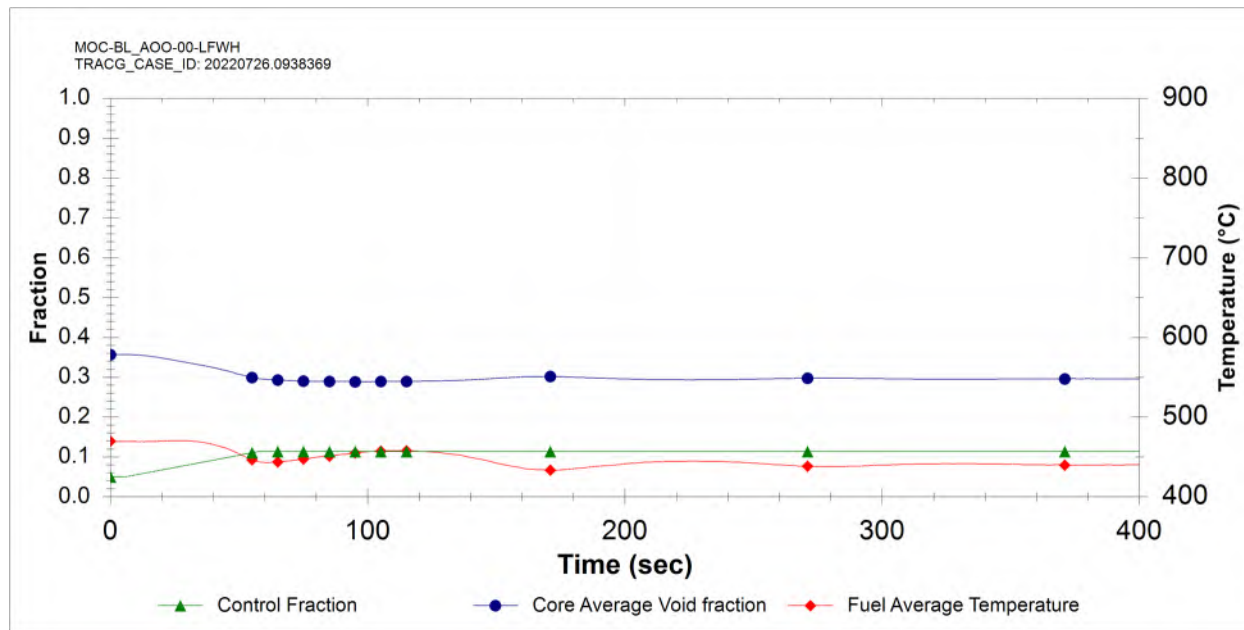


Figure 15.5-9: Loss of Feedwater Heating (AOO)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

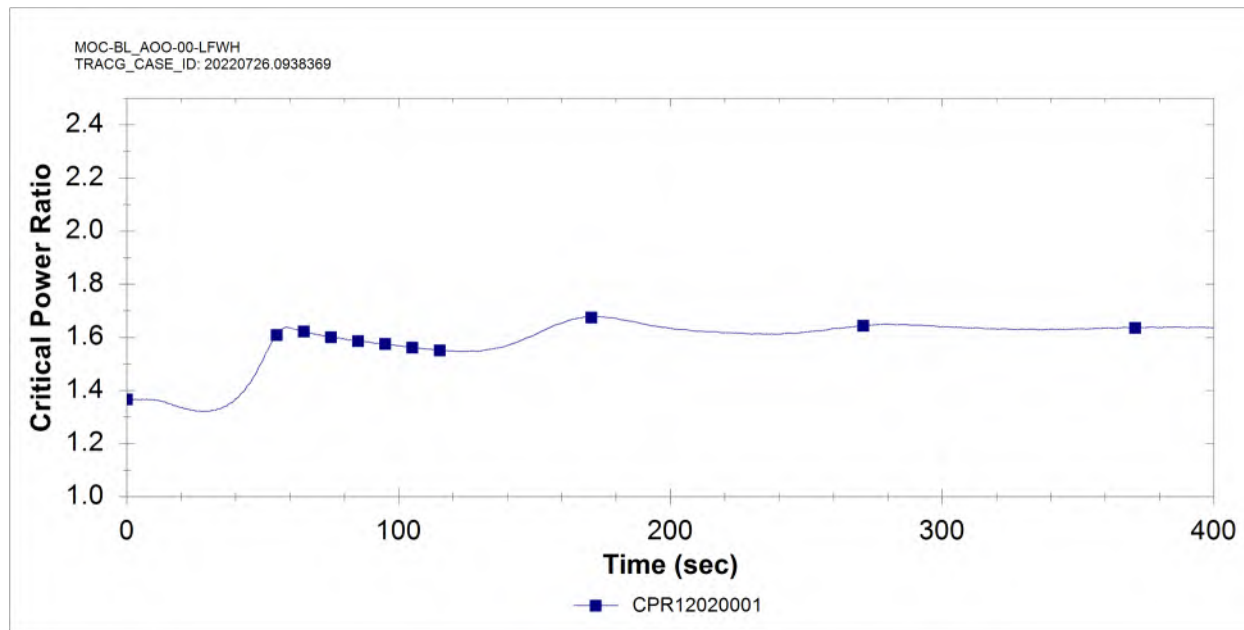


Figure 15.5-10: Loss of Feedwater Heating (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

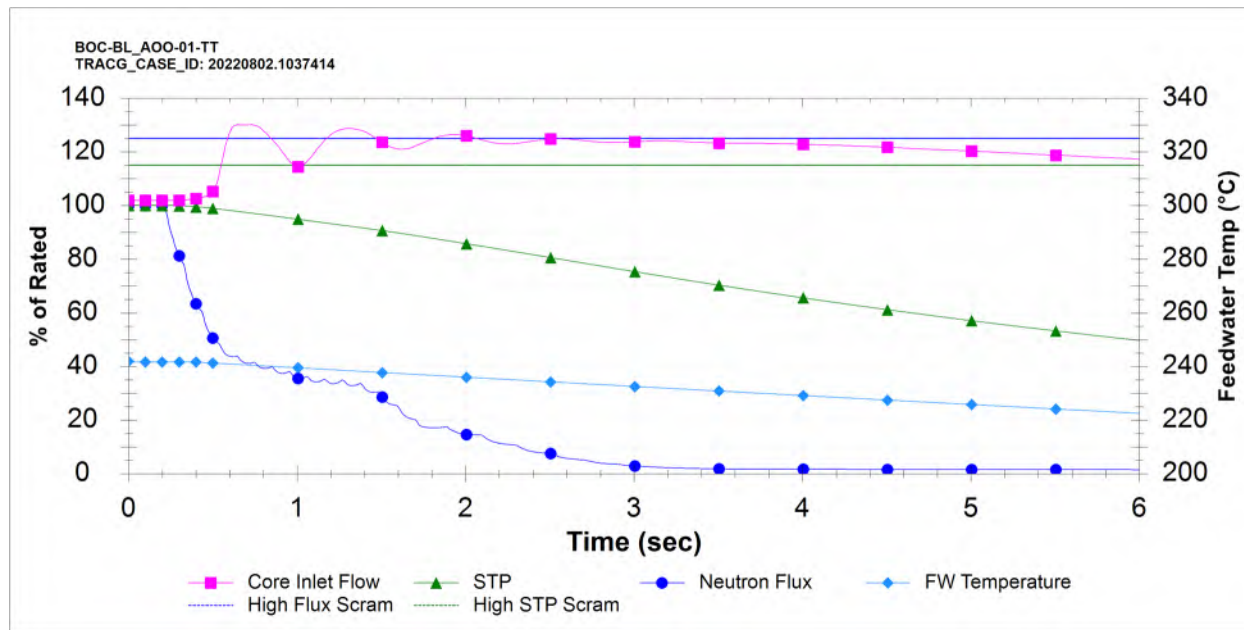


Figure 15.5-11: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

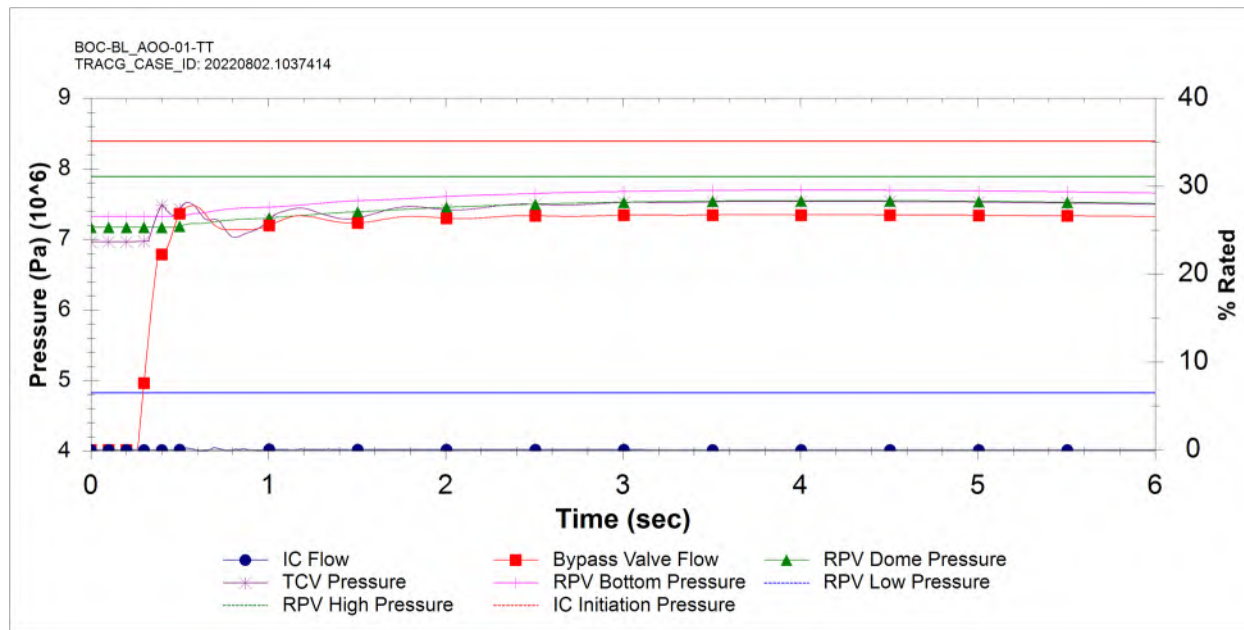


Figure 15.5-12: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

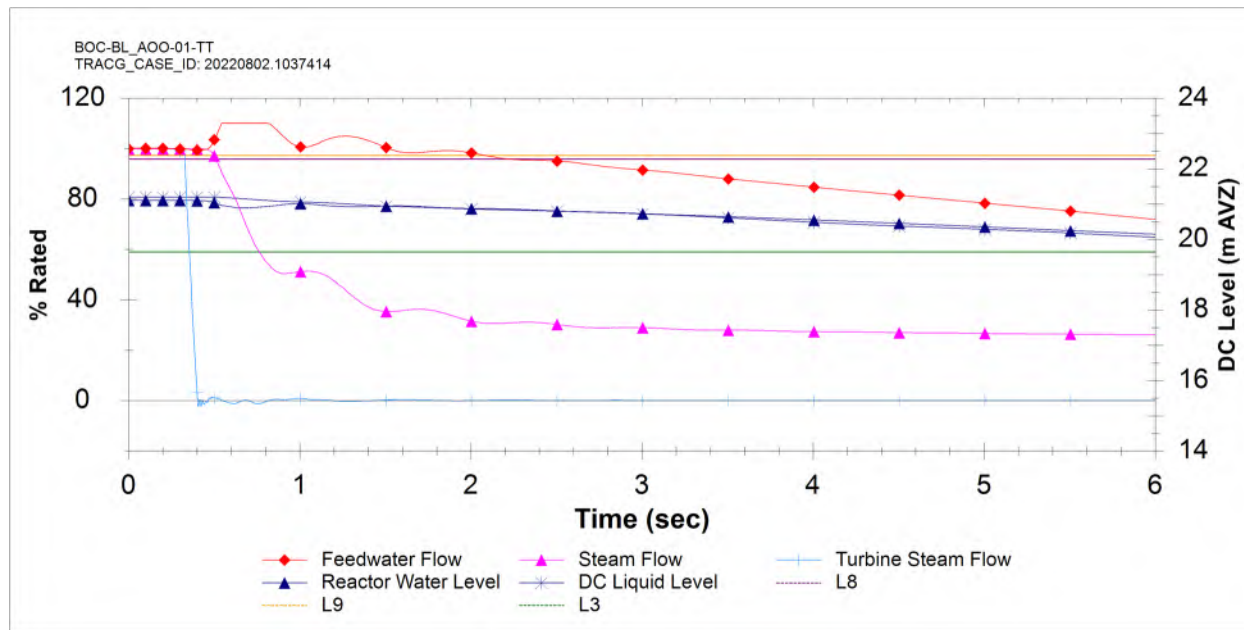


Figure 15.5-13: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

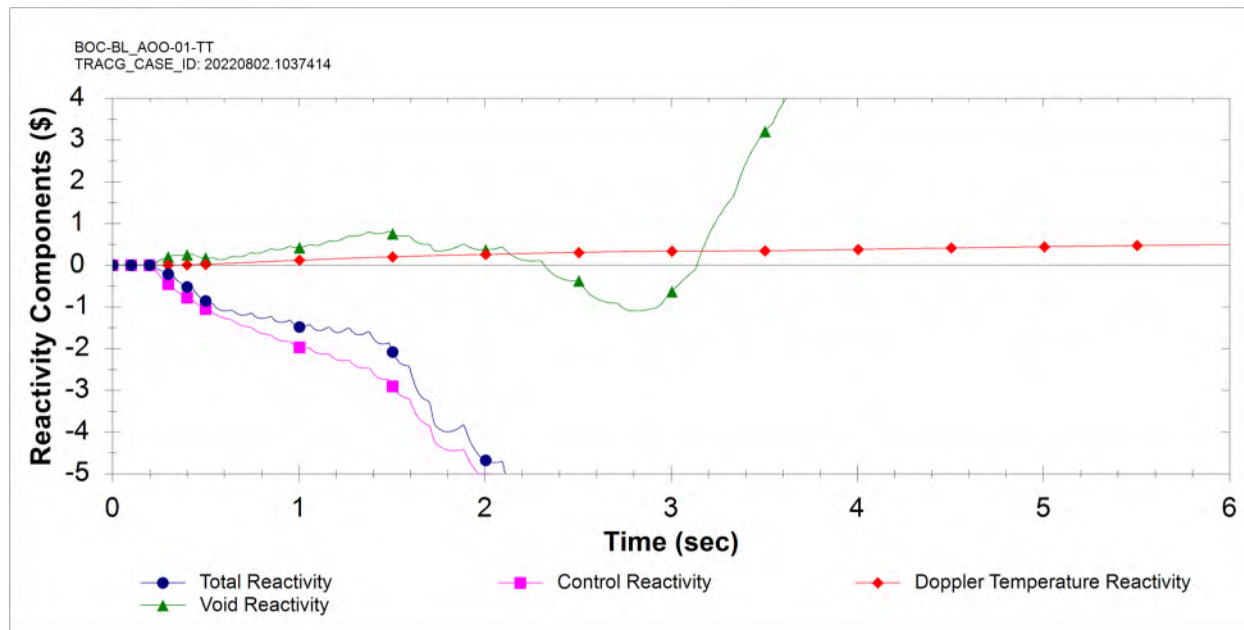


Figure 15.5-14: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

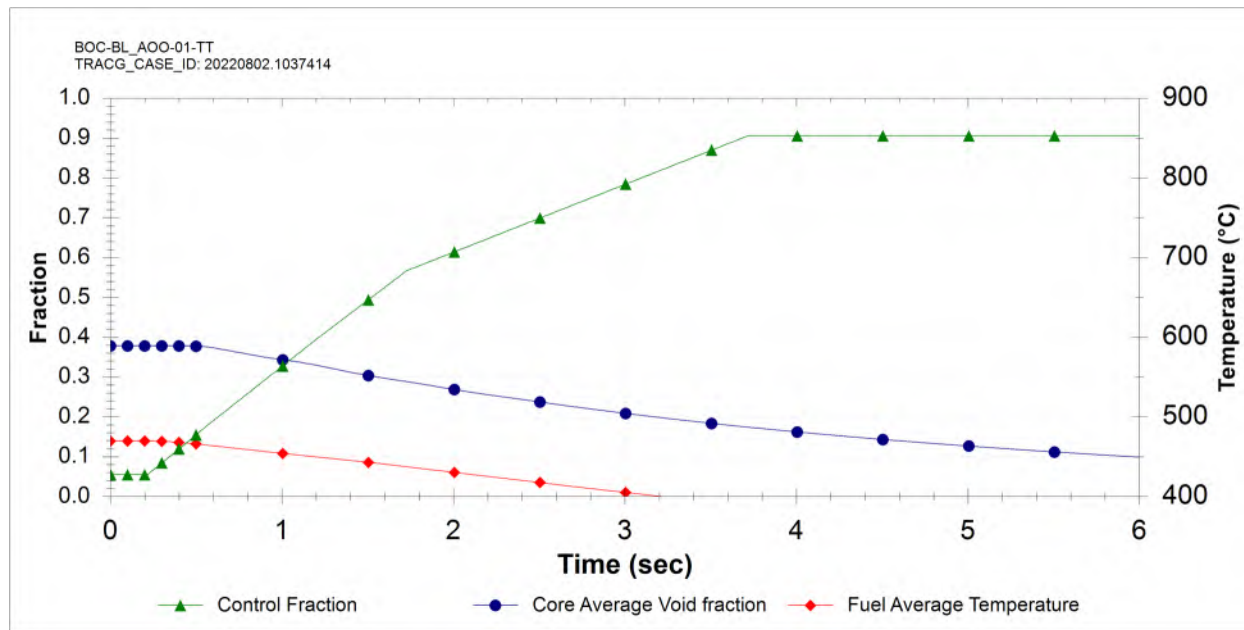


Figure 15.5-15: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

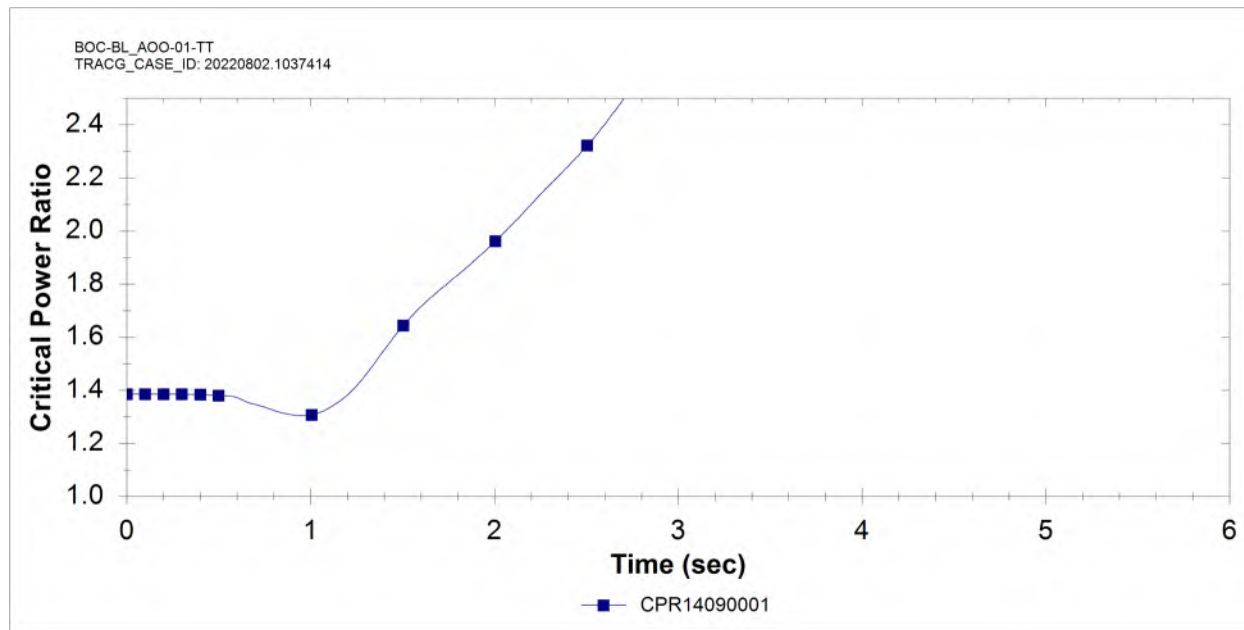


Figure 15.5-16: Turbine Trip (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

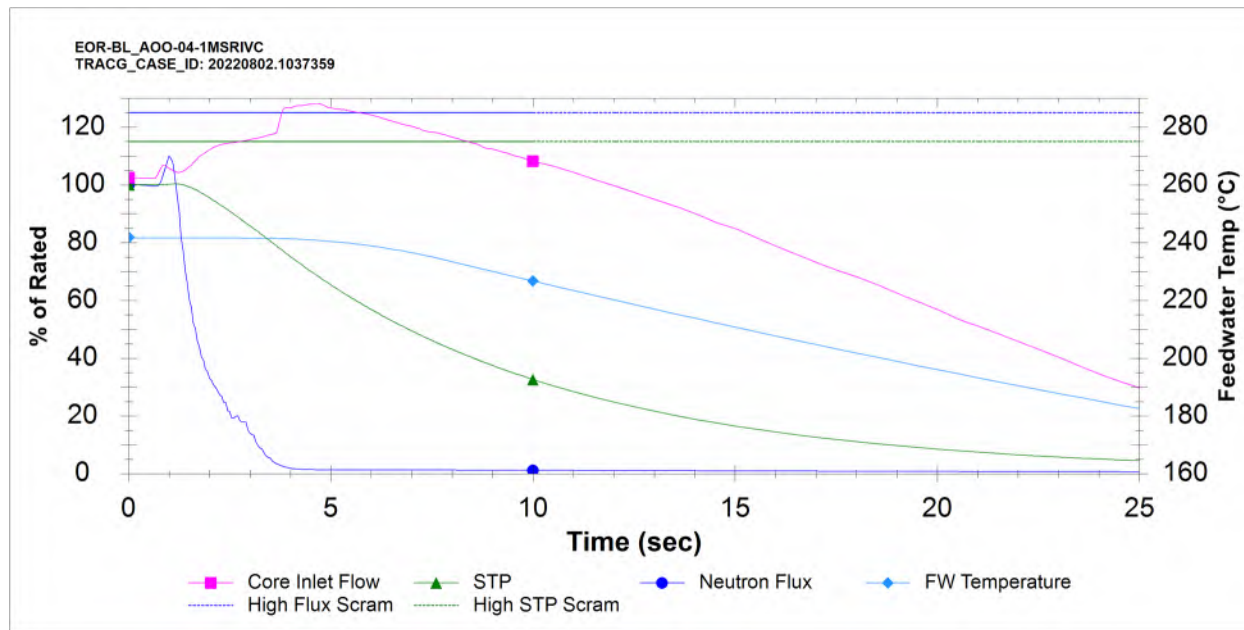


Figure 15.5-17: Closure of One Main Steam Reactor Isolation Valve (AOO)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

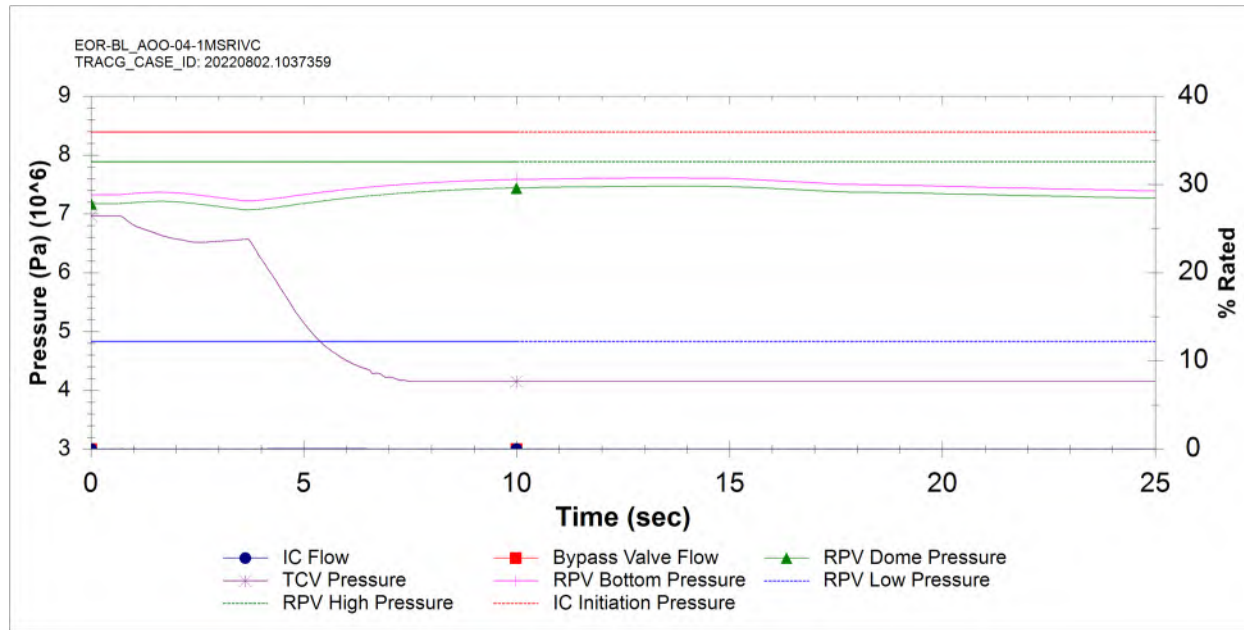


Figure 15.5-18: Closure of One Main Steam Reactor Isolation Valve (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

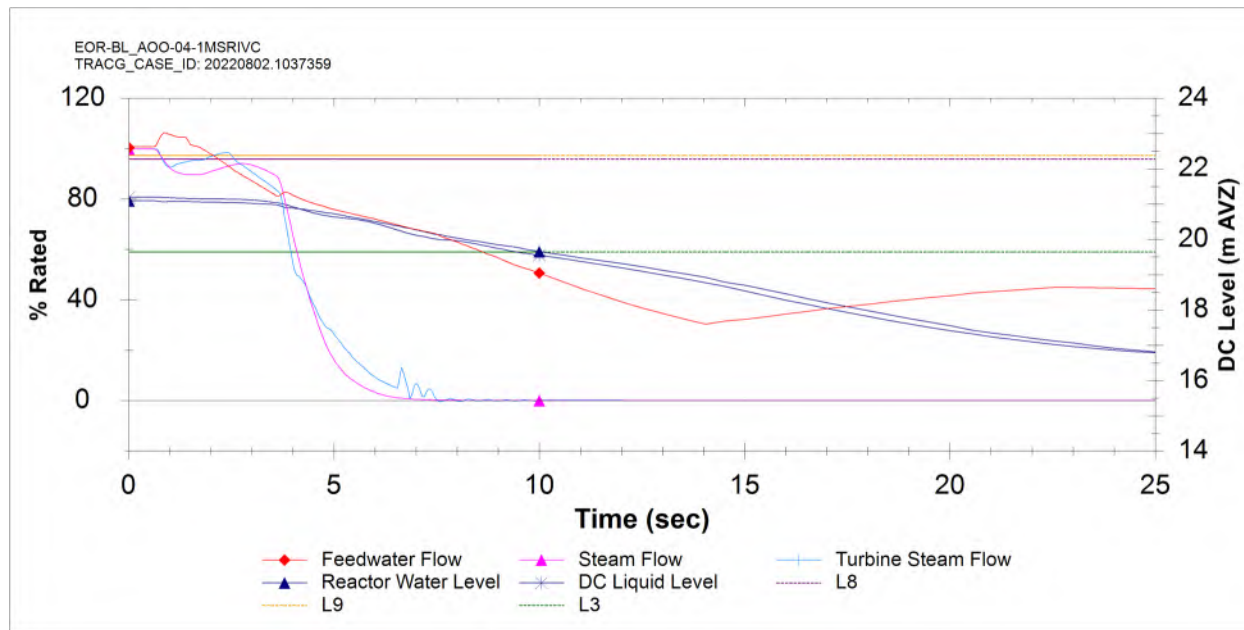
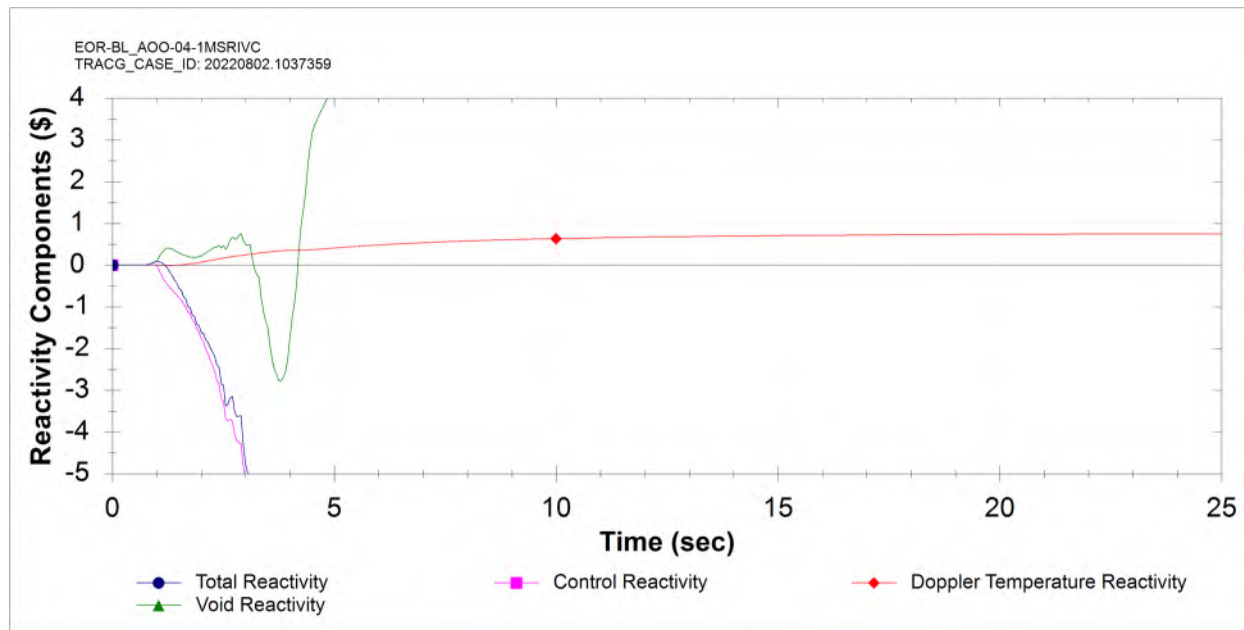


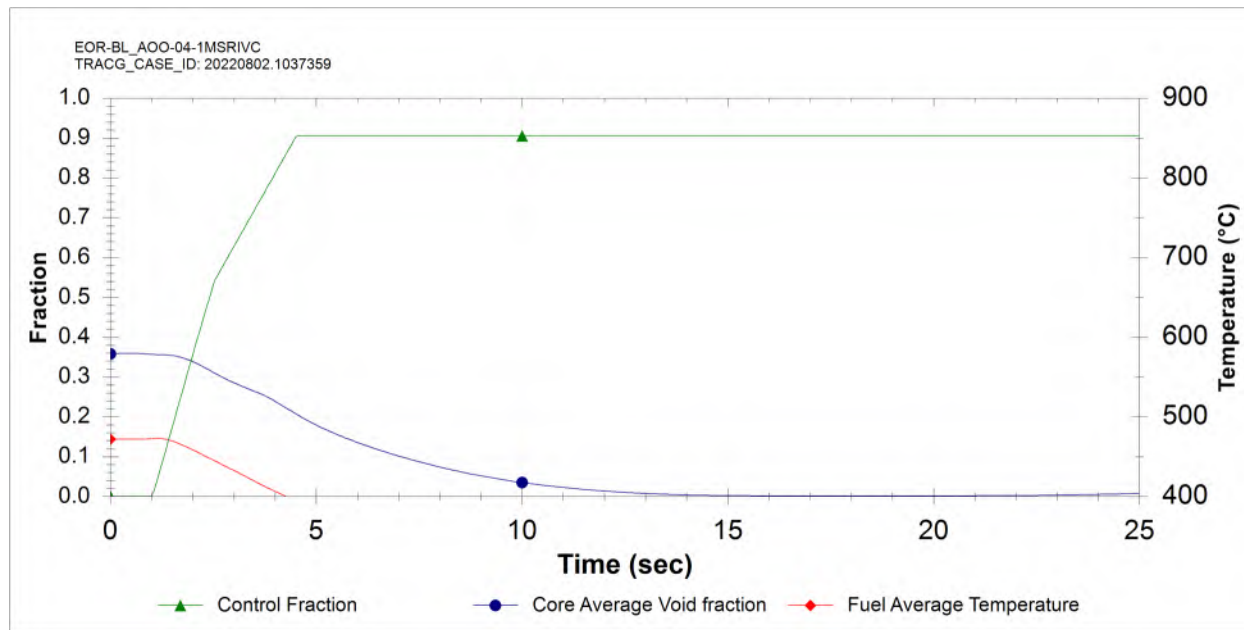
Figure 15.5-19: Closure of One Main Steam Reactor Isolation Valve (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



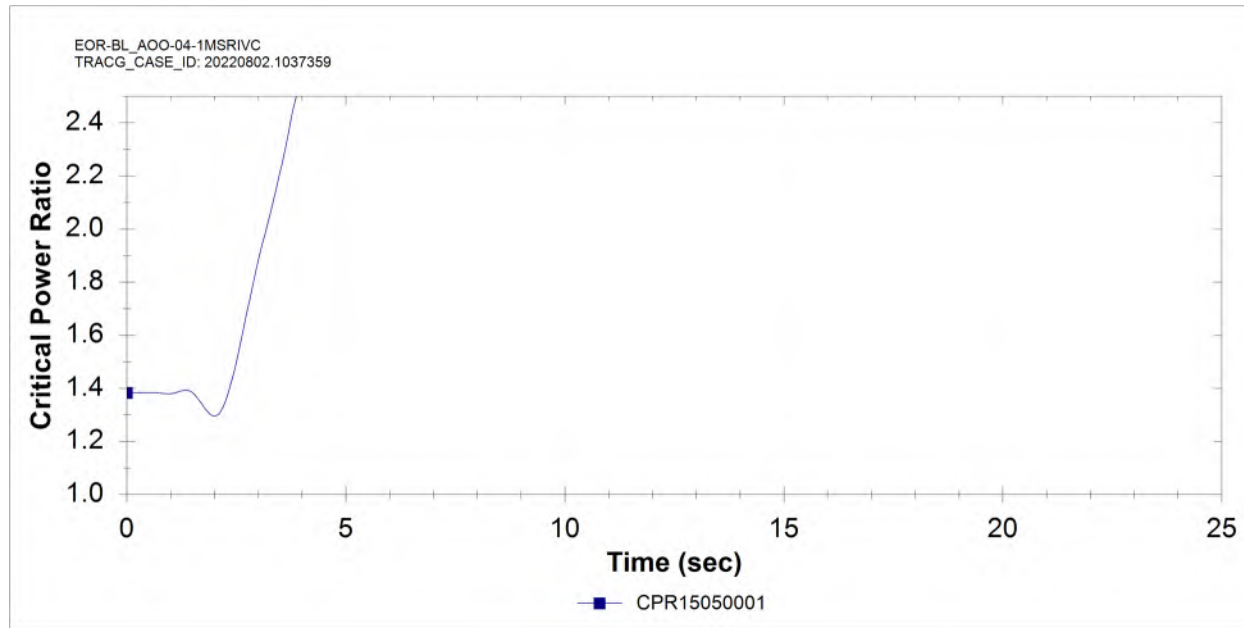
**Figure 15.5-20: Closure of One Main Steam Reactor Isolation Valve (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



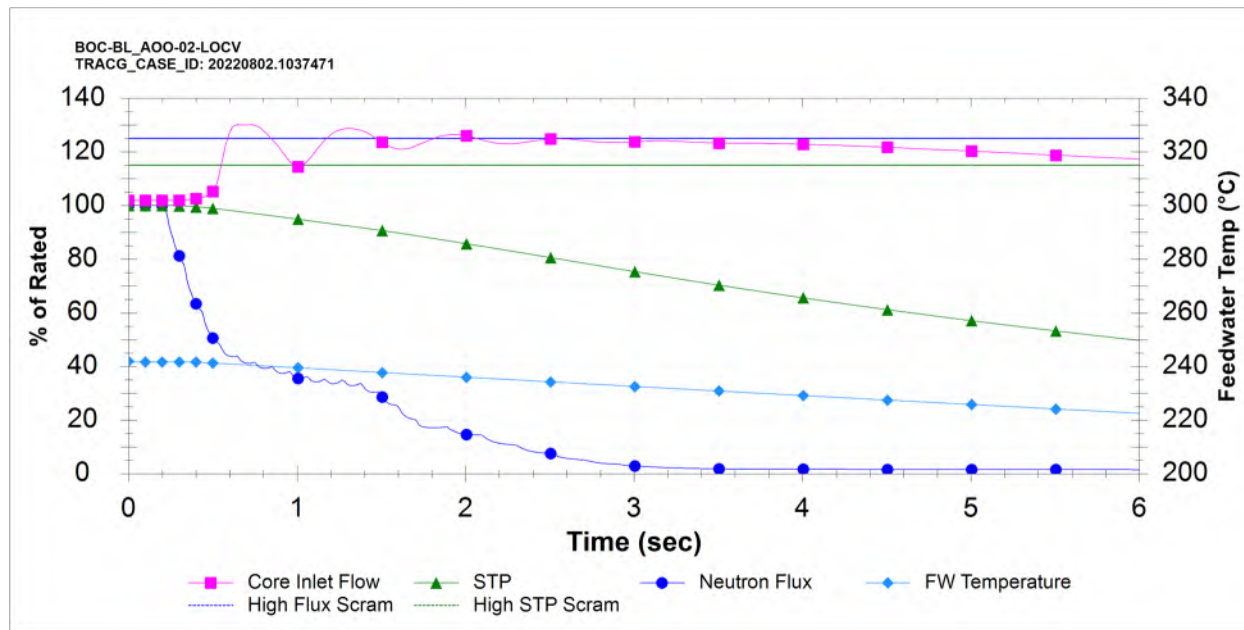
**Figure 15.5-21: Closure of One Main Steam Reactor Isolation Valve (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-22: Closure of One Main Steam Reactor Isolation Valve (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-23: Loss of Condenser Vacuum (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

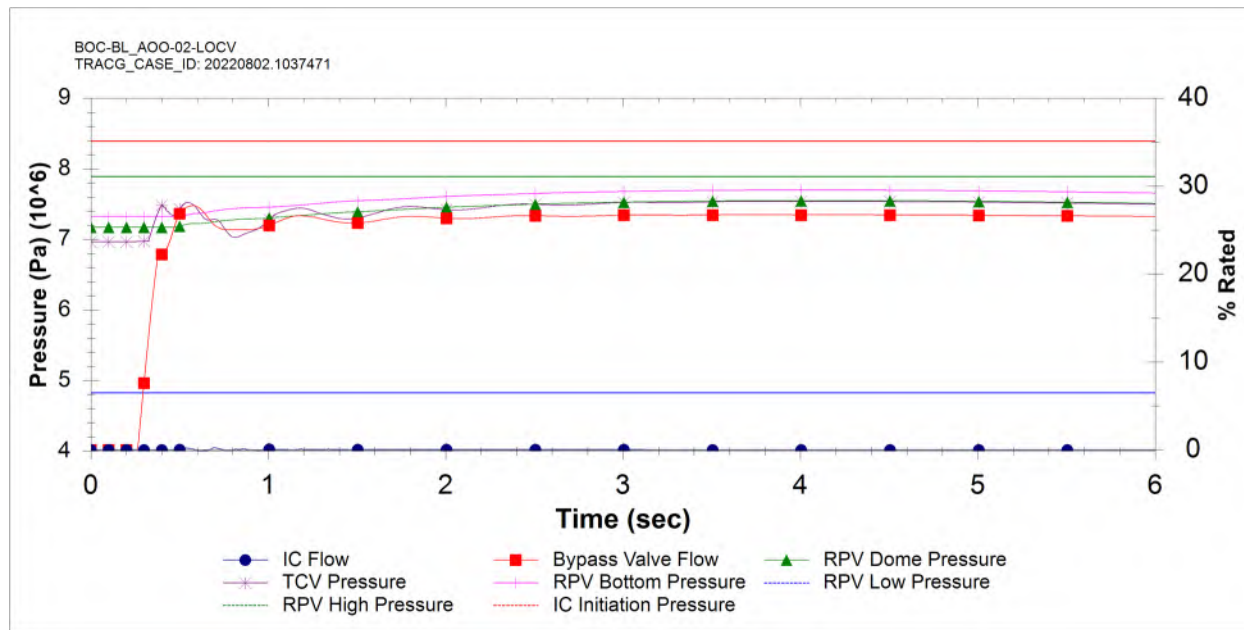


Figure 15.5-23a: Loss of Condenser Vacuum (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

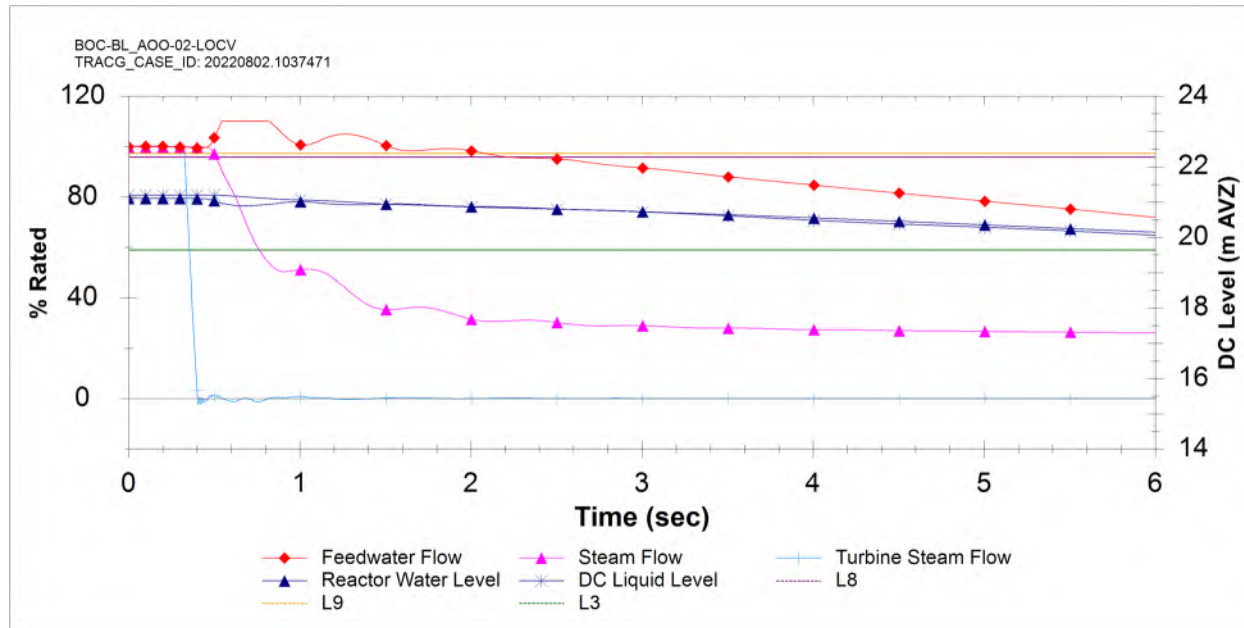
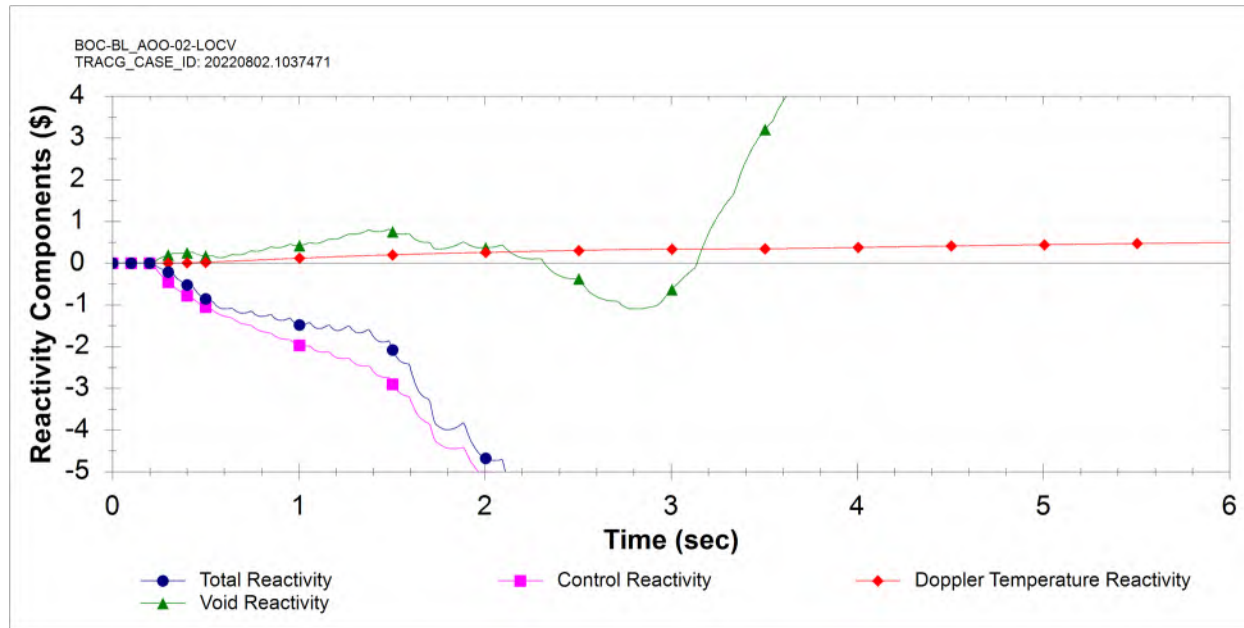


Figure 15.5-24: Loss of Condenser Vacuum (AOO)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-25: Loss of Condenser Vacuum (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

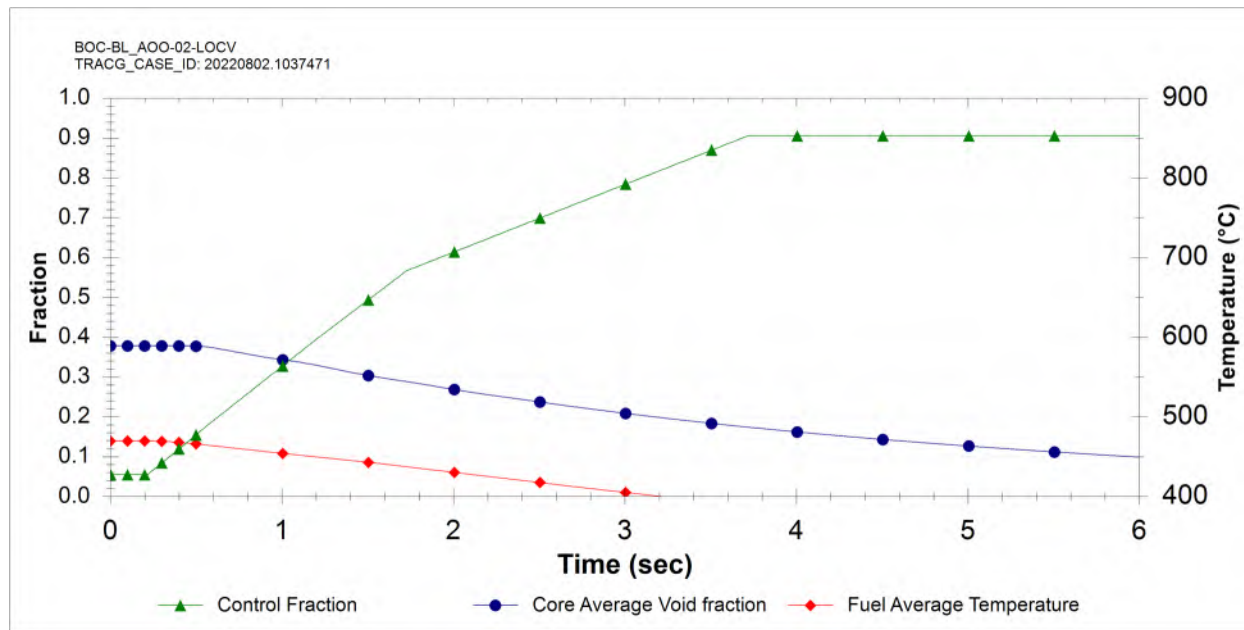


Figure 15.5-26: Loss of Condenser Vacuum (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

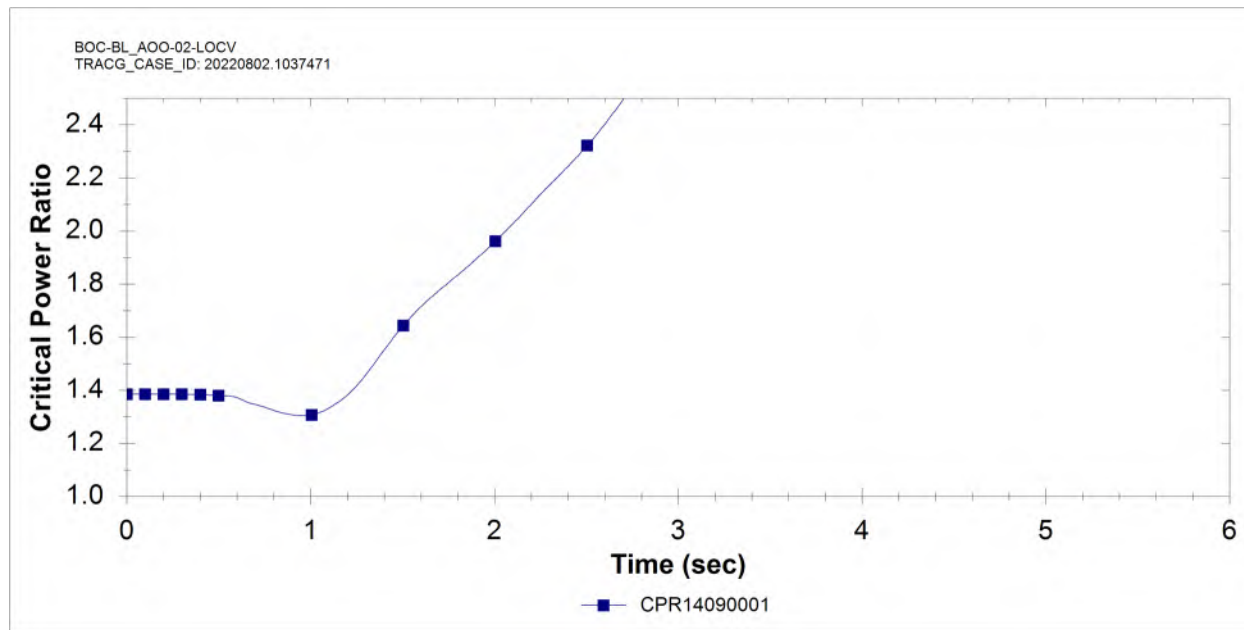


Figure 15.5-27: Loss of Condenser Vacuum (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

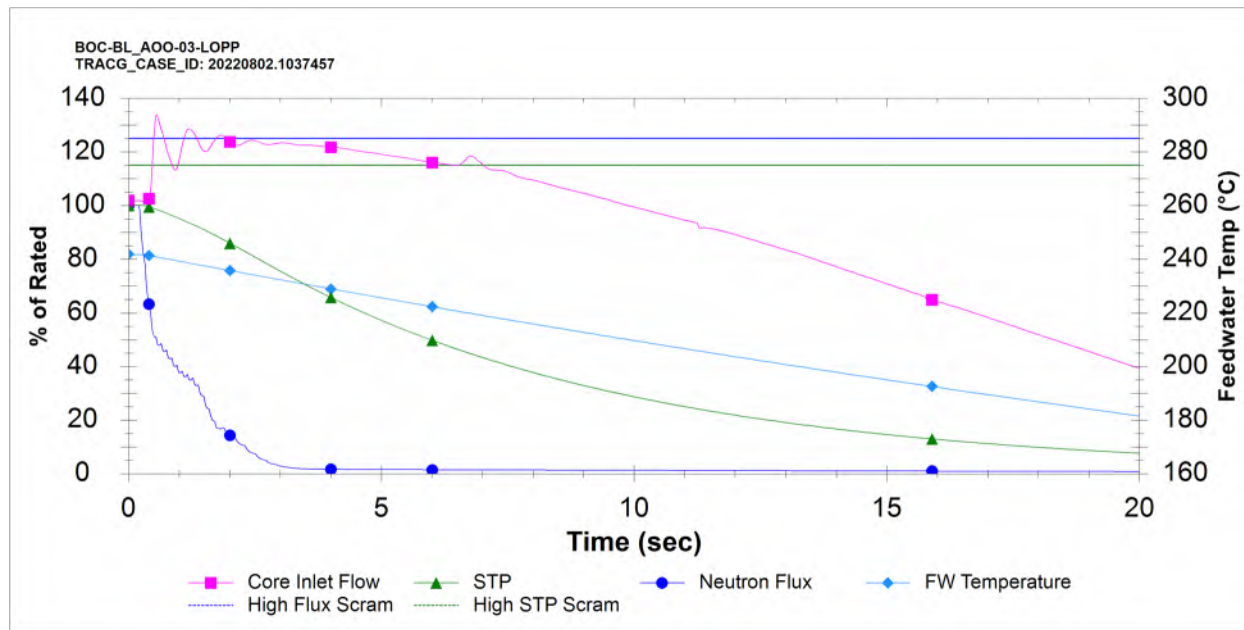


Figure 15.5-28: Loss of Preferred Power (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

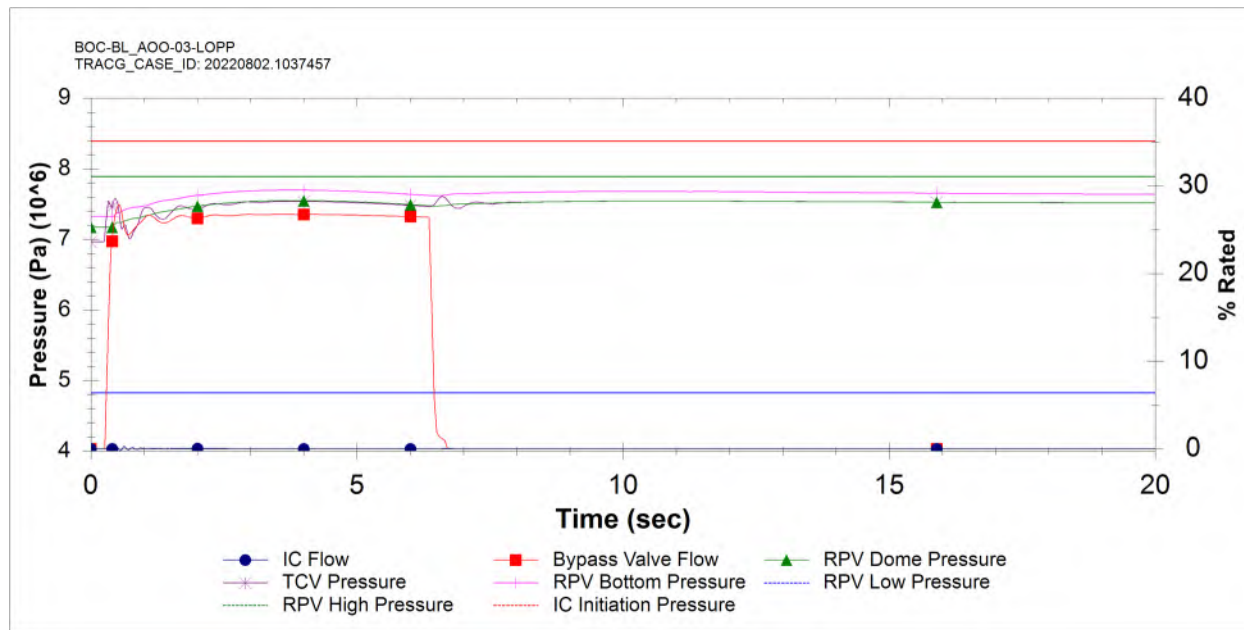


Figure 15.5-29: Loss of Preferred Power (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

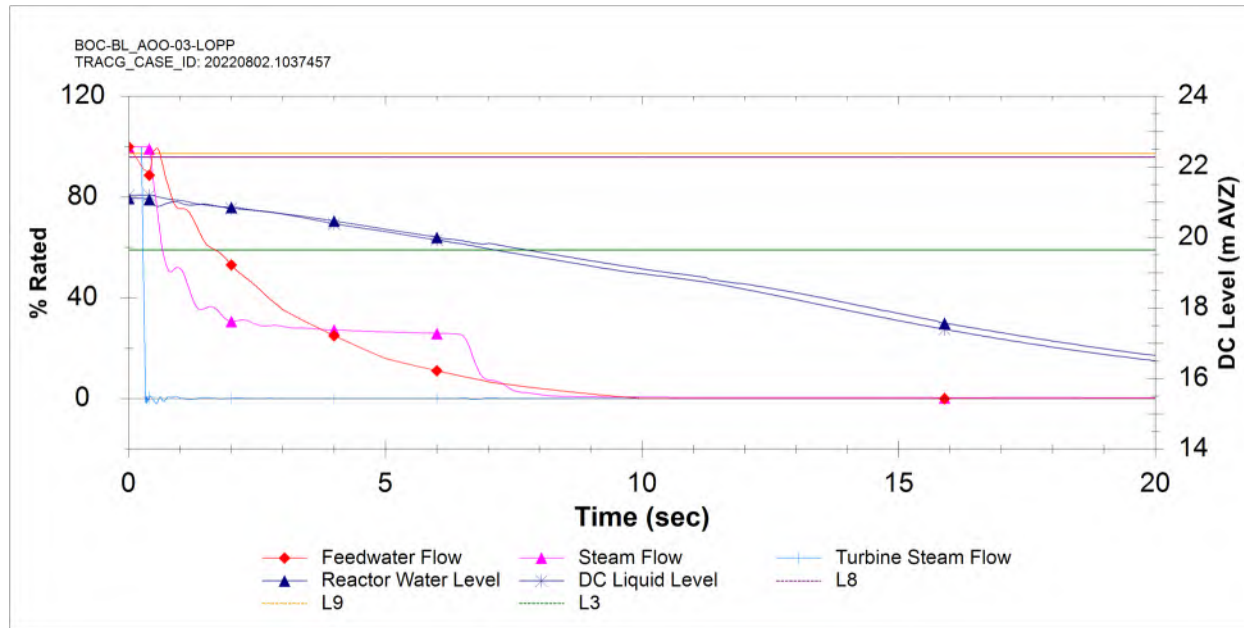


Figure 15.5-30: Loss of Preferred Power (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

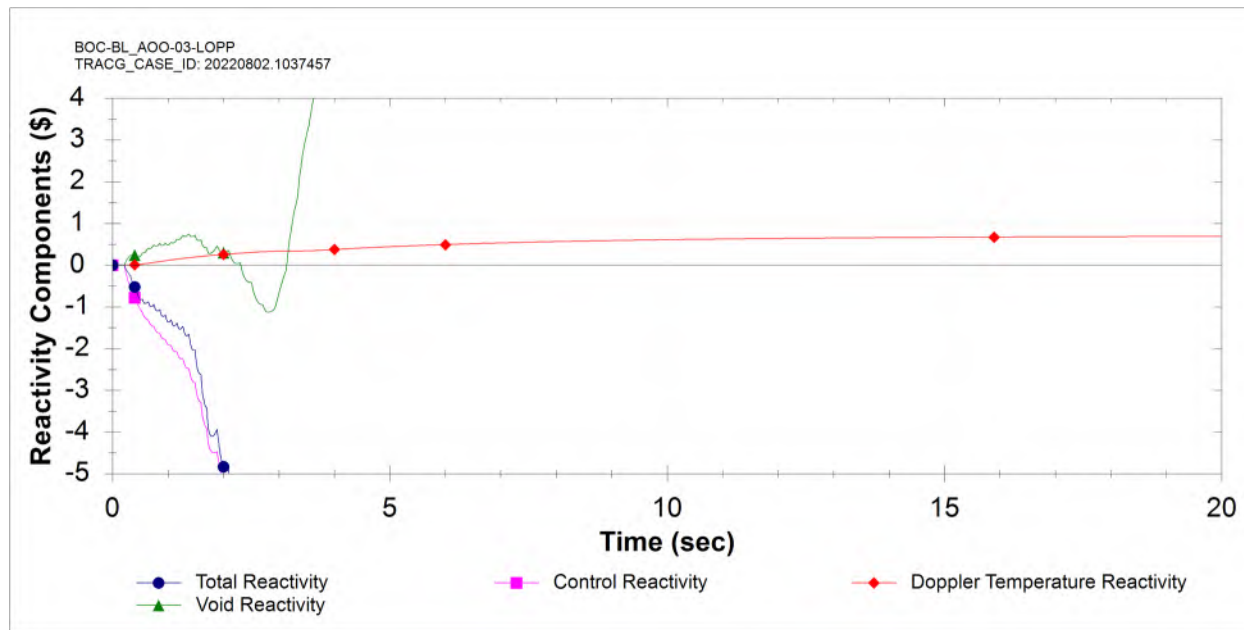


Figure 15.5-31: Loss of Preferred Power (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

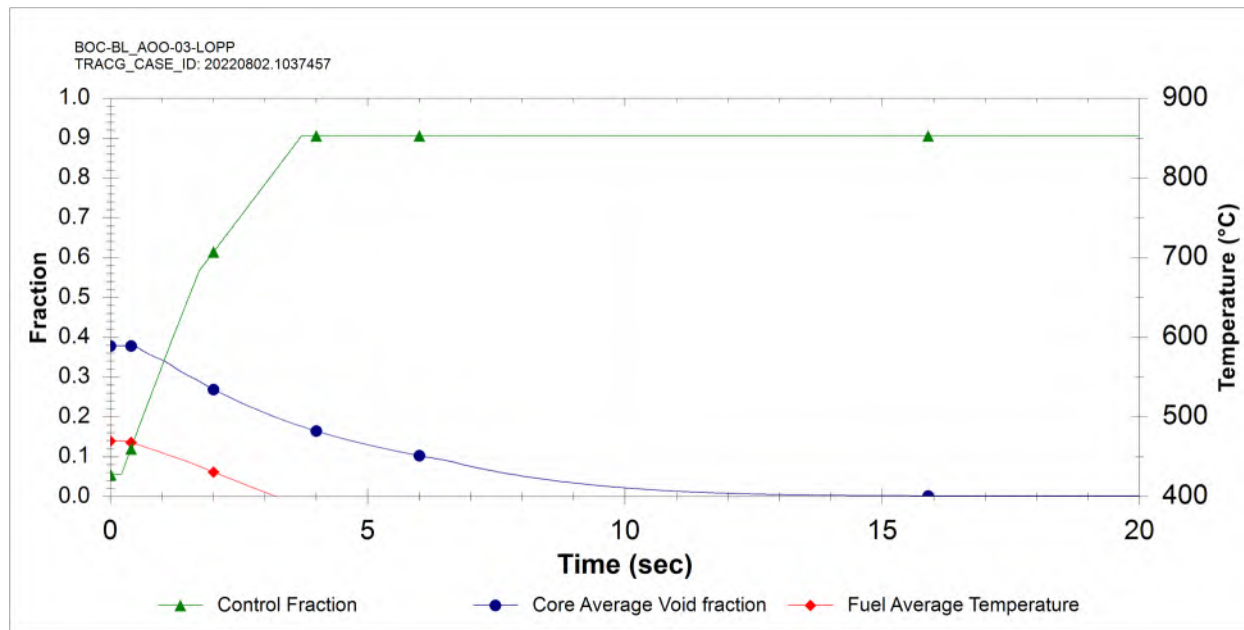
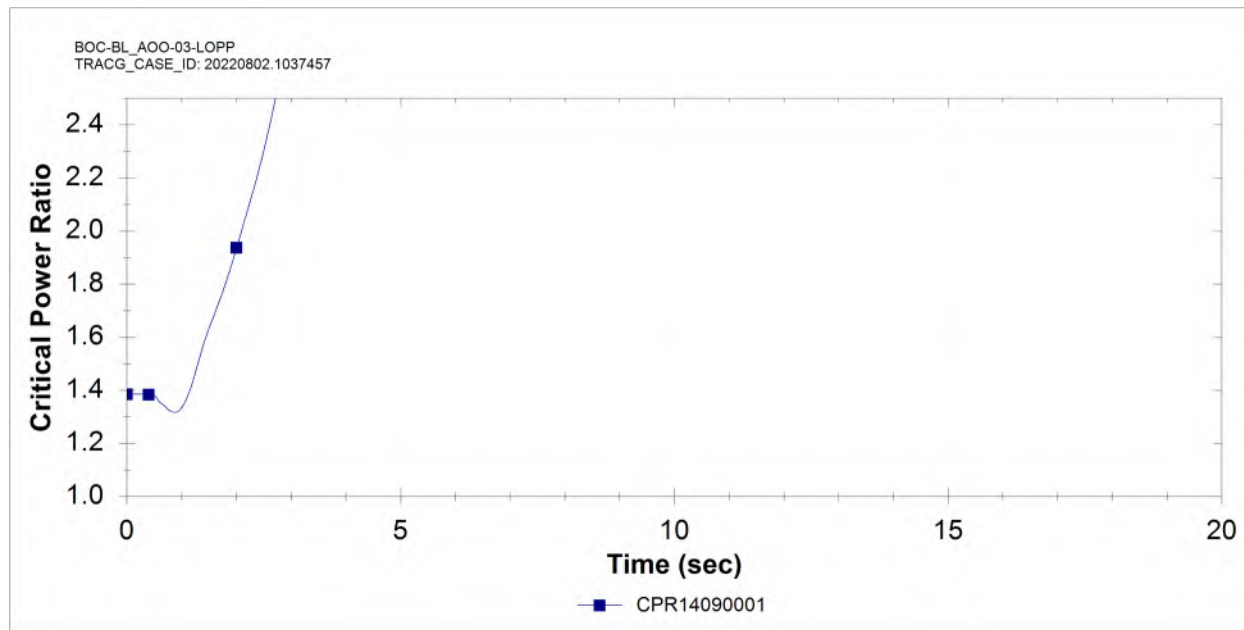


Figure 15.5-32: Loss of Preferred Power (AOO)

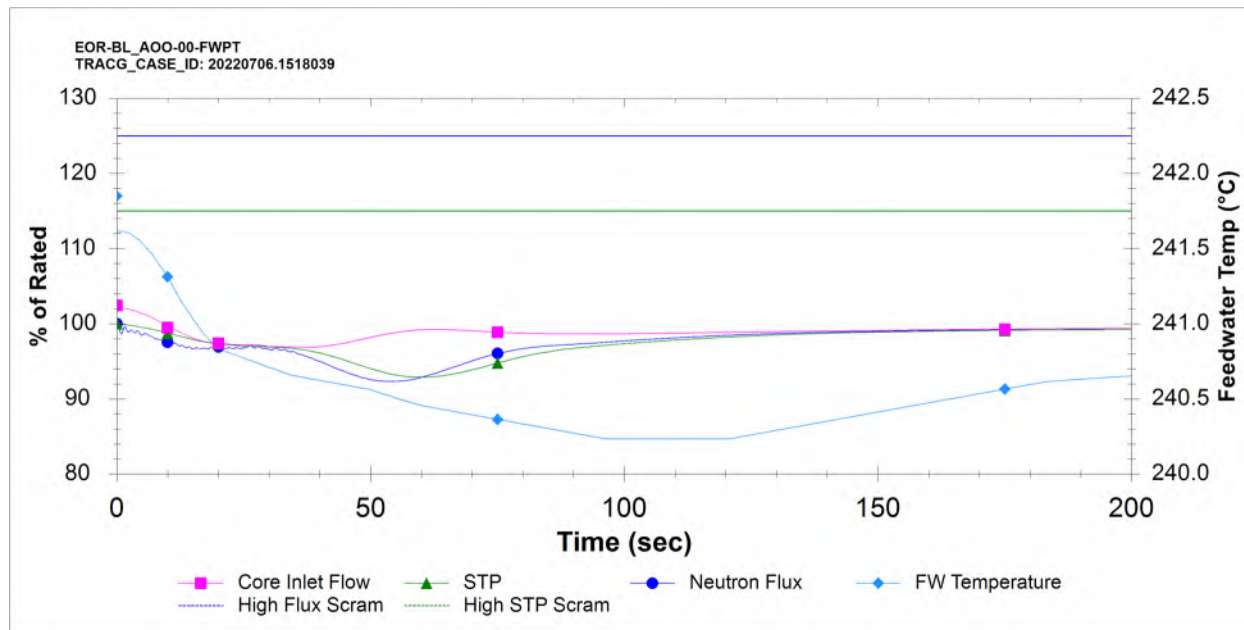


NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



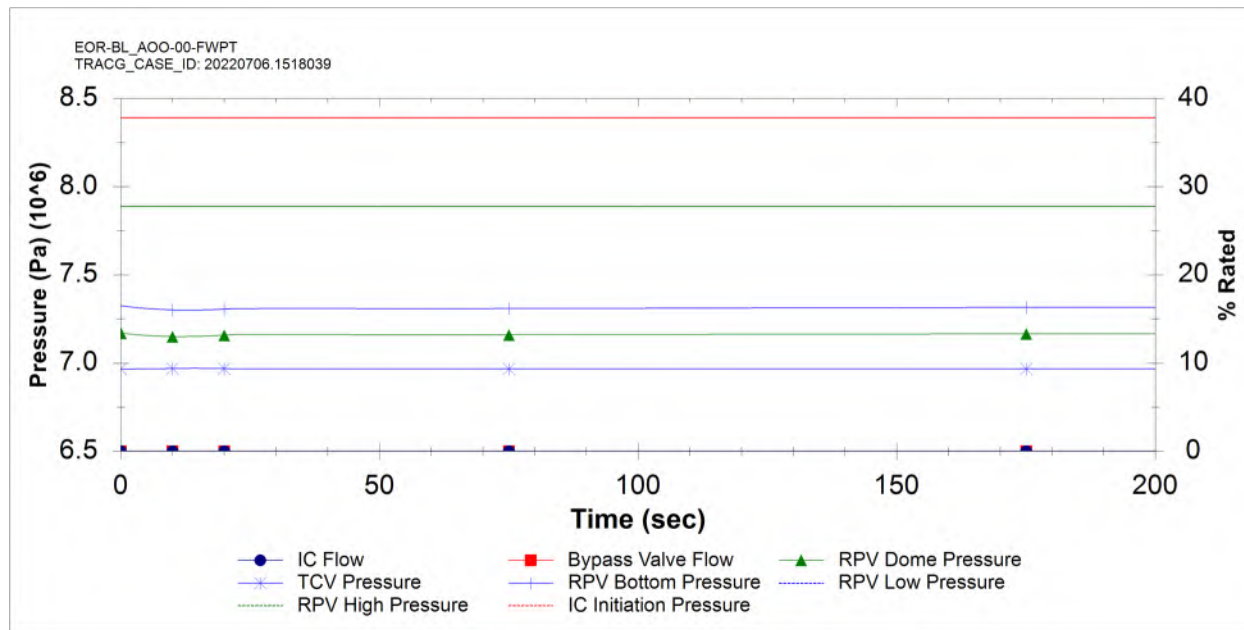
**Figure 15.5-33: Loss of Preferred Power (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



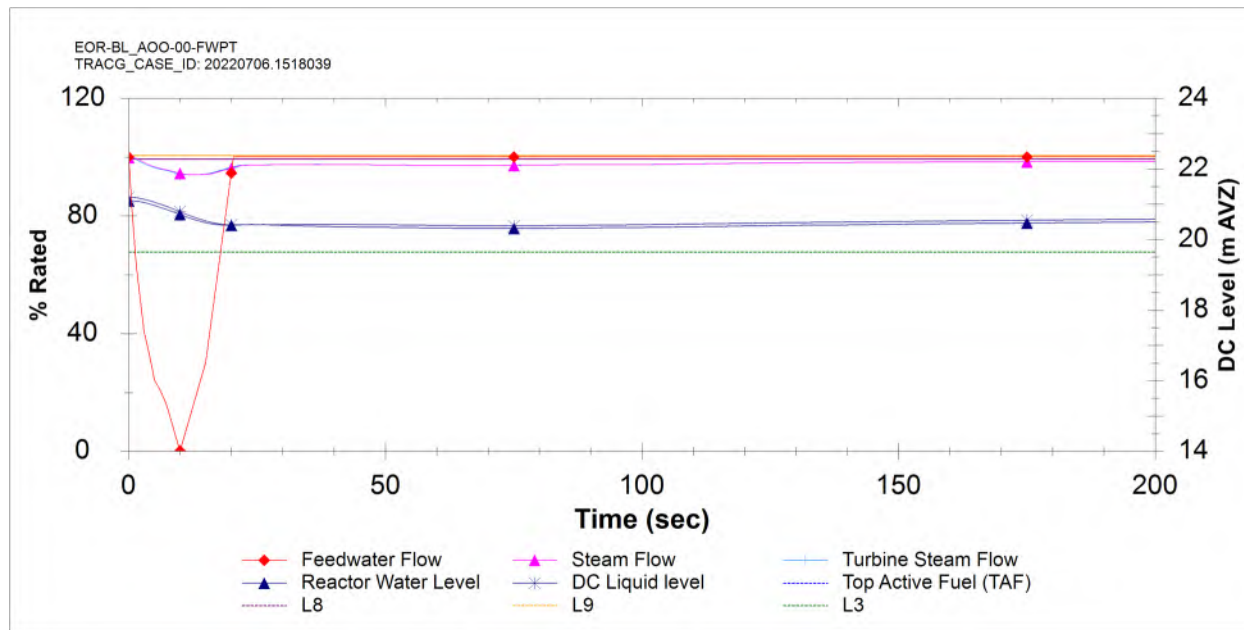
**Figure 15.5-34: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



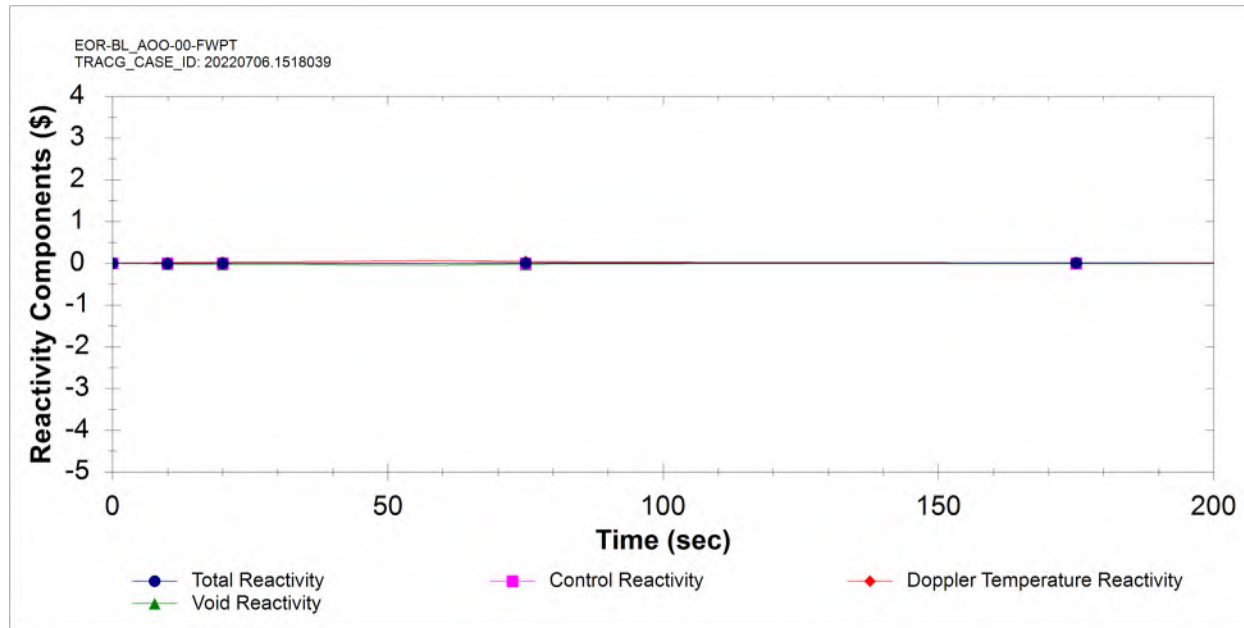
**Figure 15.5-35: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



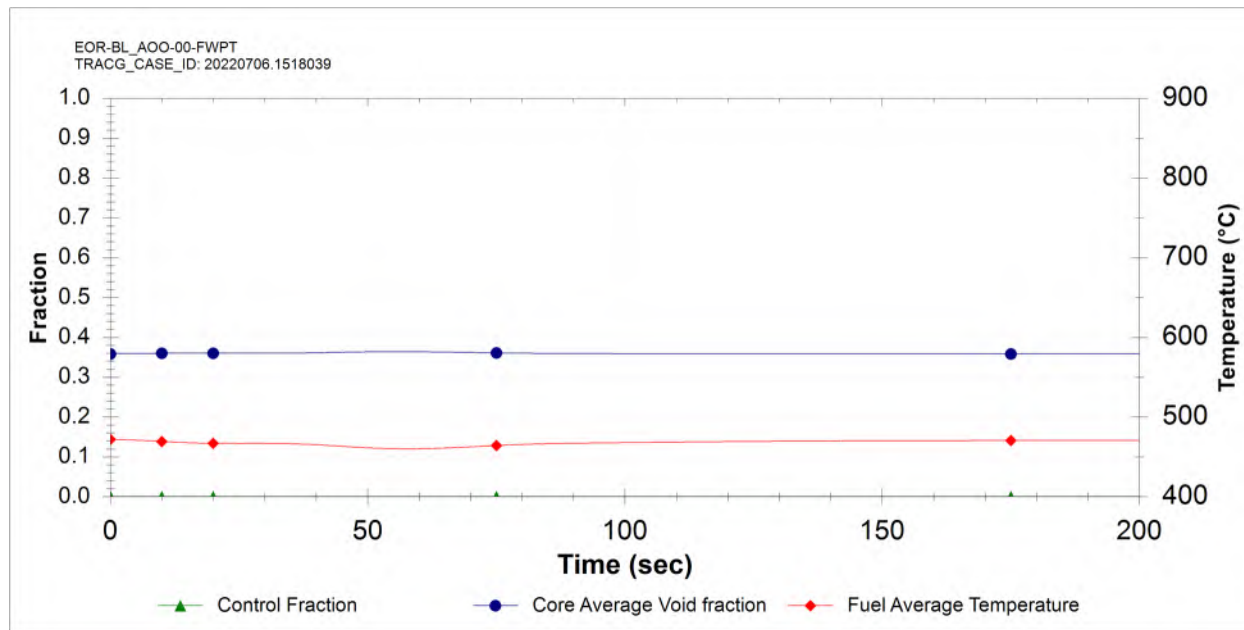
**Figure 15.5-36: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



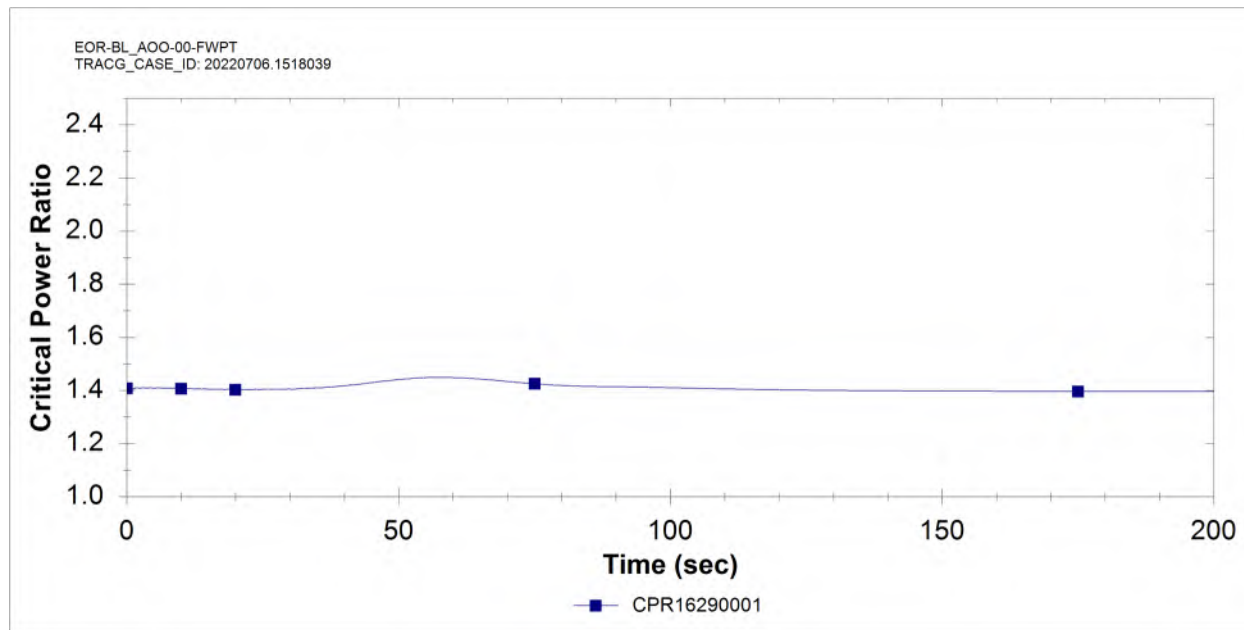
**Figure 15.5-37: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



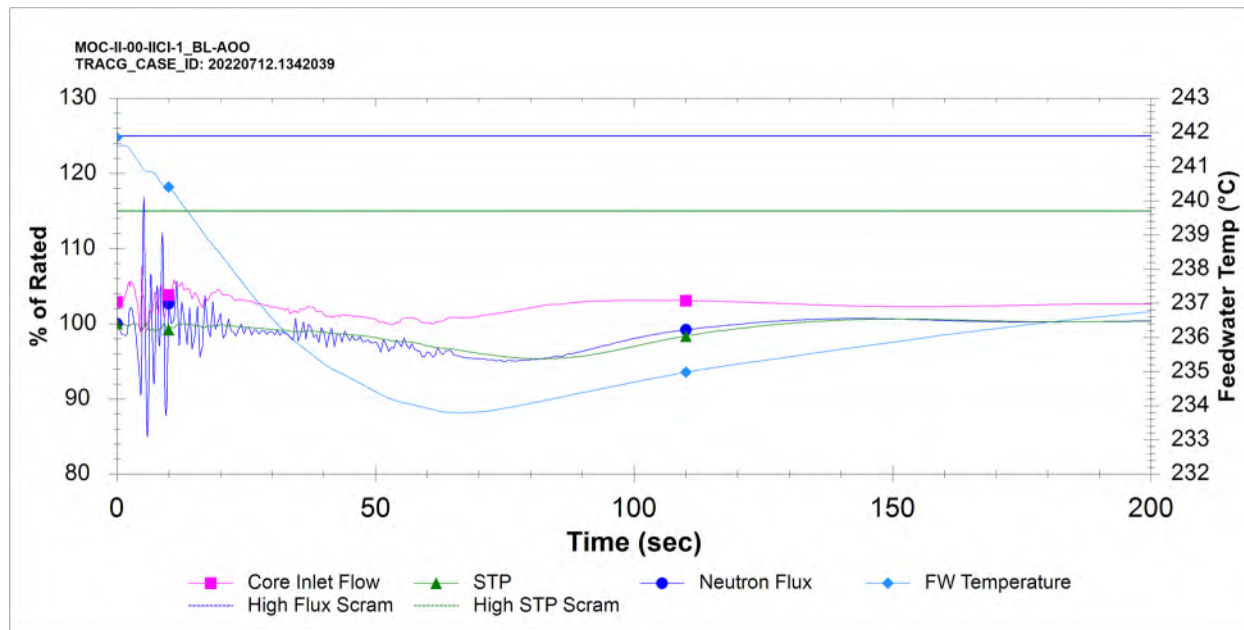
**Figure 15.5-38: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-39: Feedwater Pump Trip – One Pump (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-40: Inadvertent Isolation Condenser Initiation – One Train (AOO)**



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

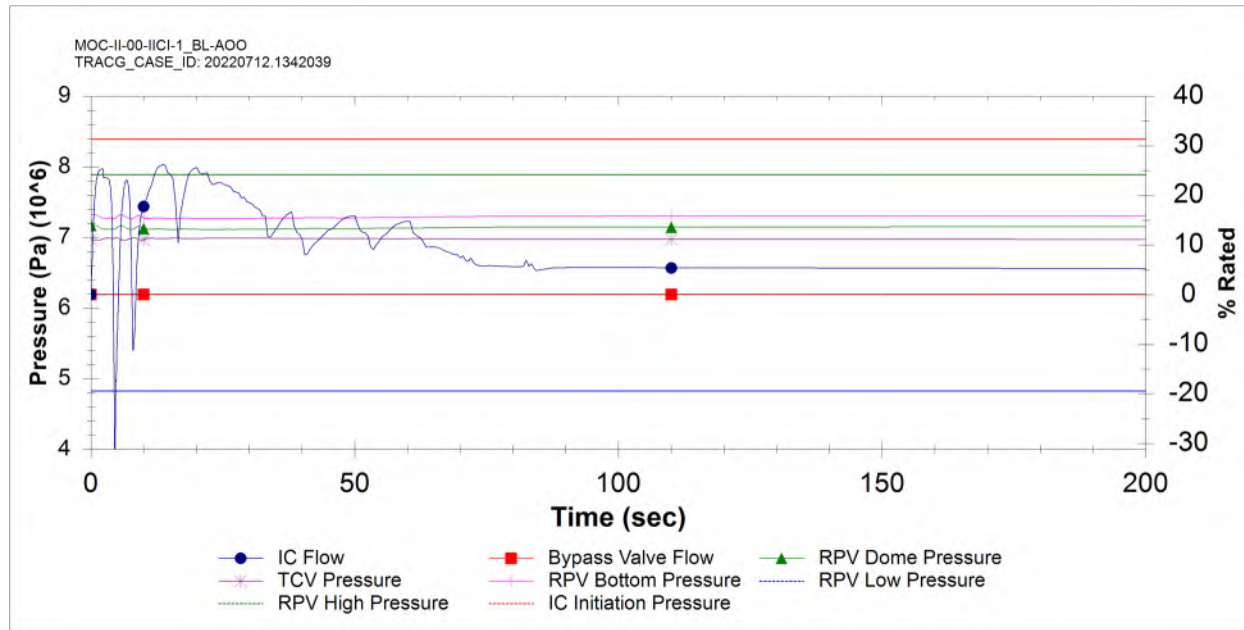


Figure 15.5-41: Inadvertent Isolation Condenser Initiation – One Train (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

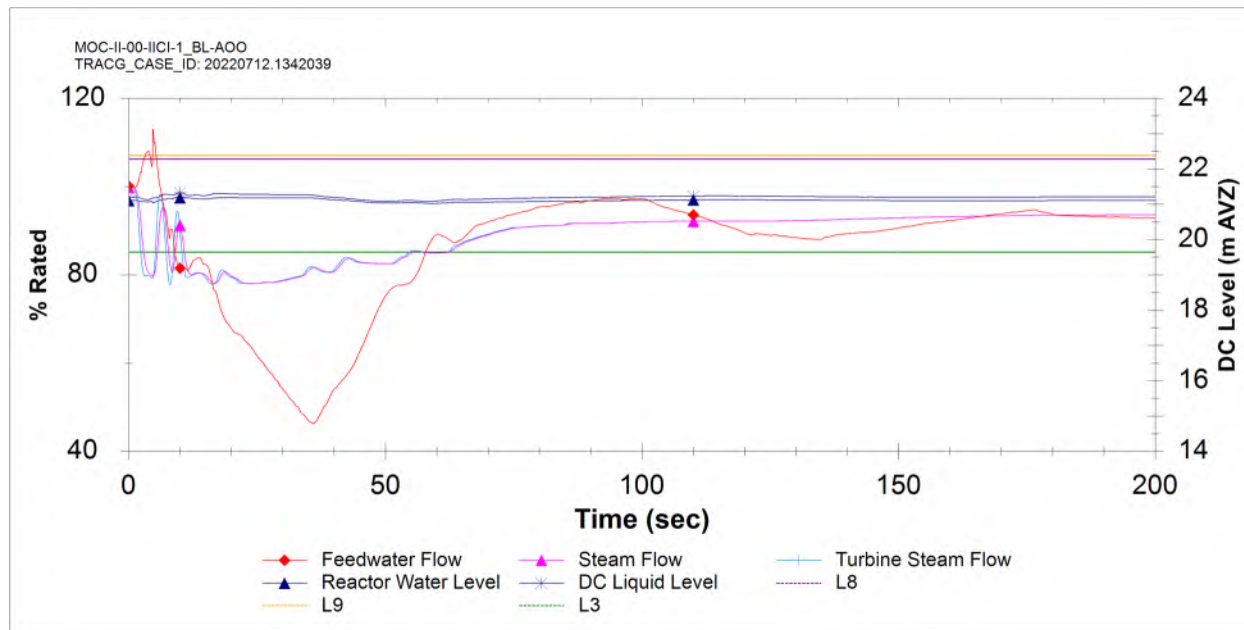
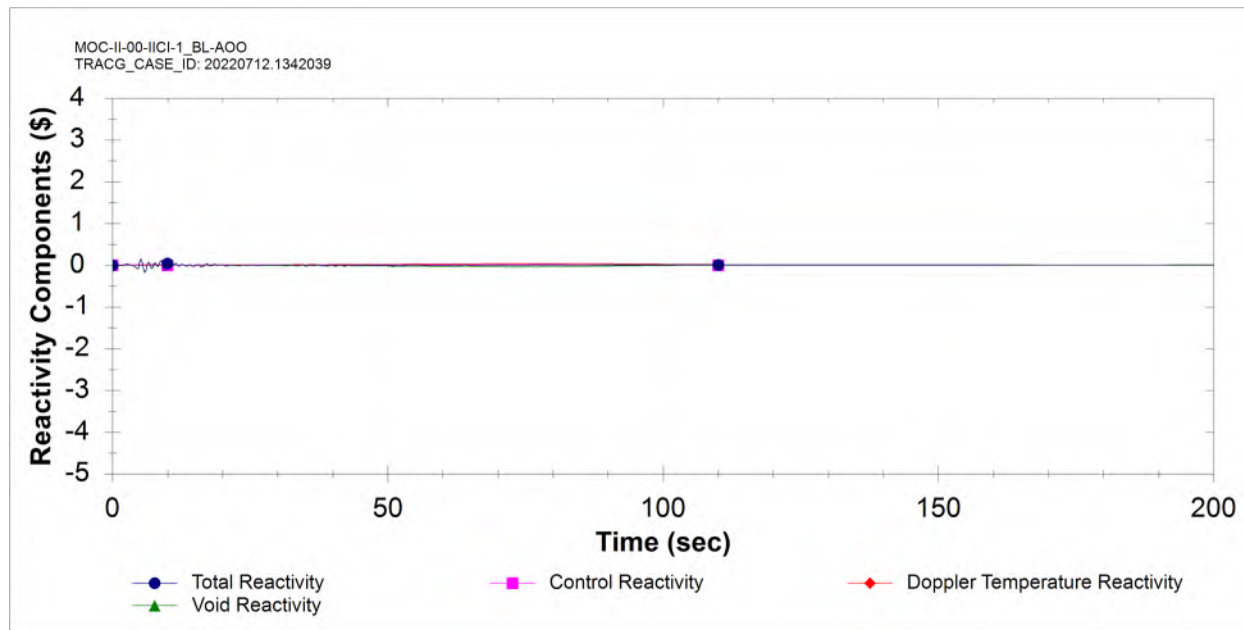


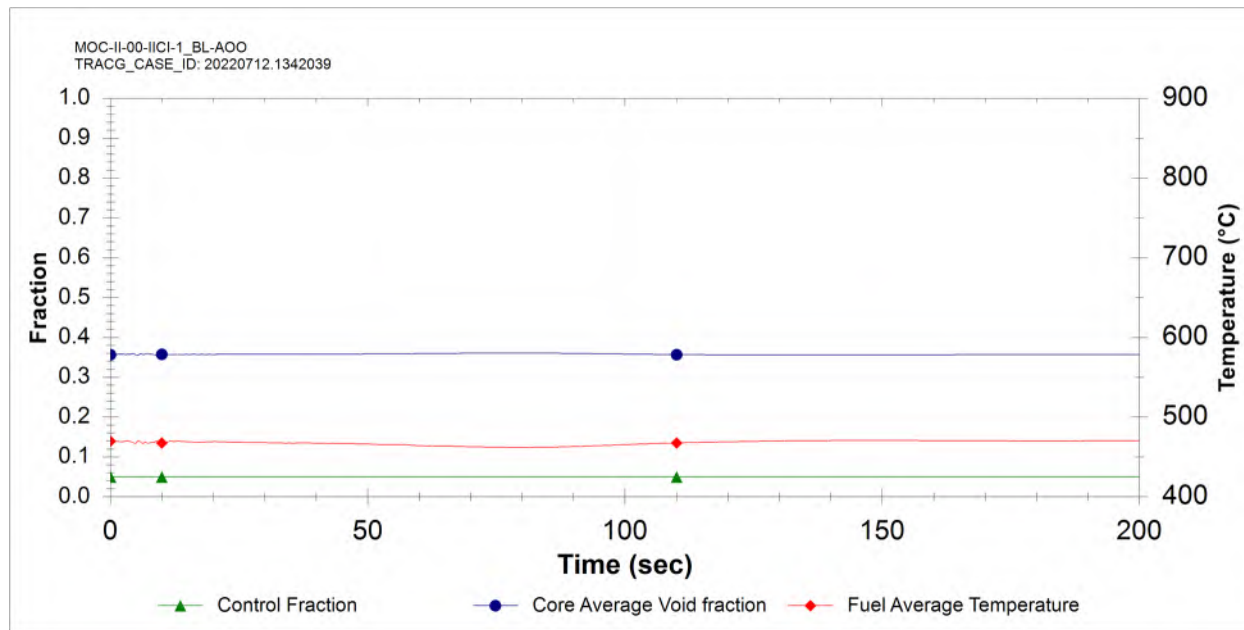
Figure 15.5-42: Inadvertent Isolation Condenser Initiation – One Train (AOO)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



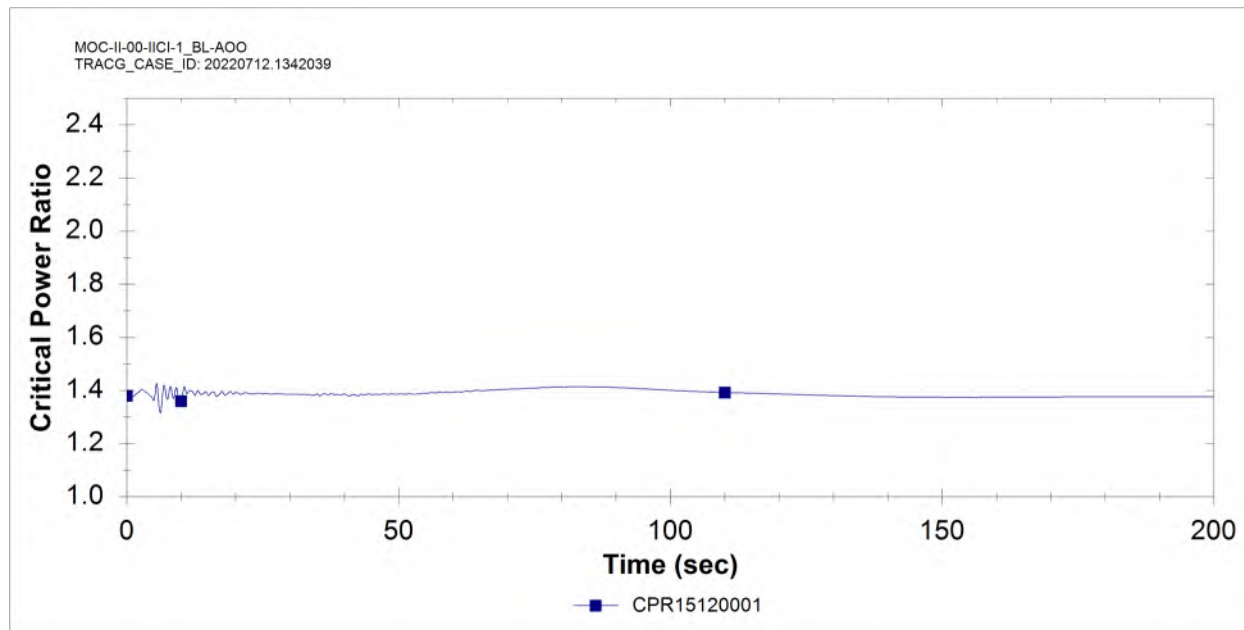
**Figure 15.5-43: Inadvertent Isolation Condenser Initiation - One Train (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-44: Inadvertent Isolation Condenser Initiation - One Train (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-45: Inadvertent Isolation Condenser Initiation - One Train (AOO)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

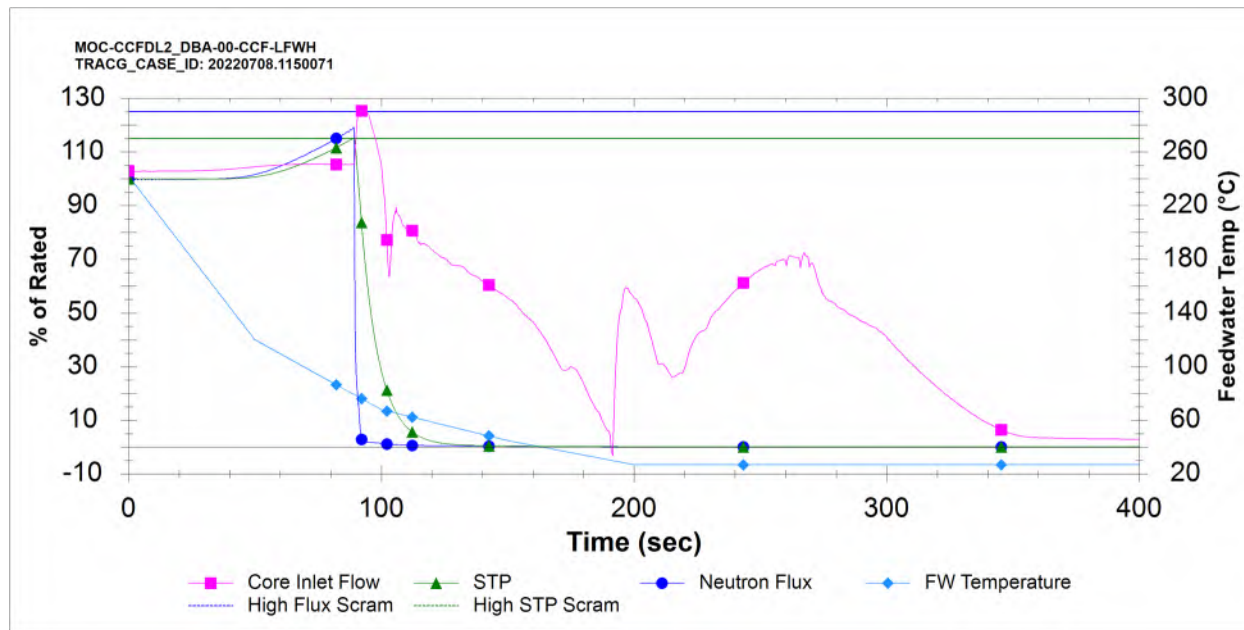


Figure 15.5-46: Loss of Feedwater Heating (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

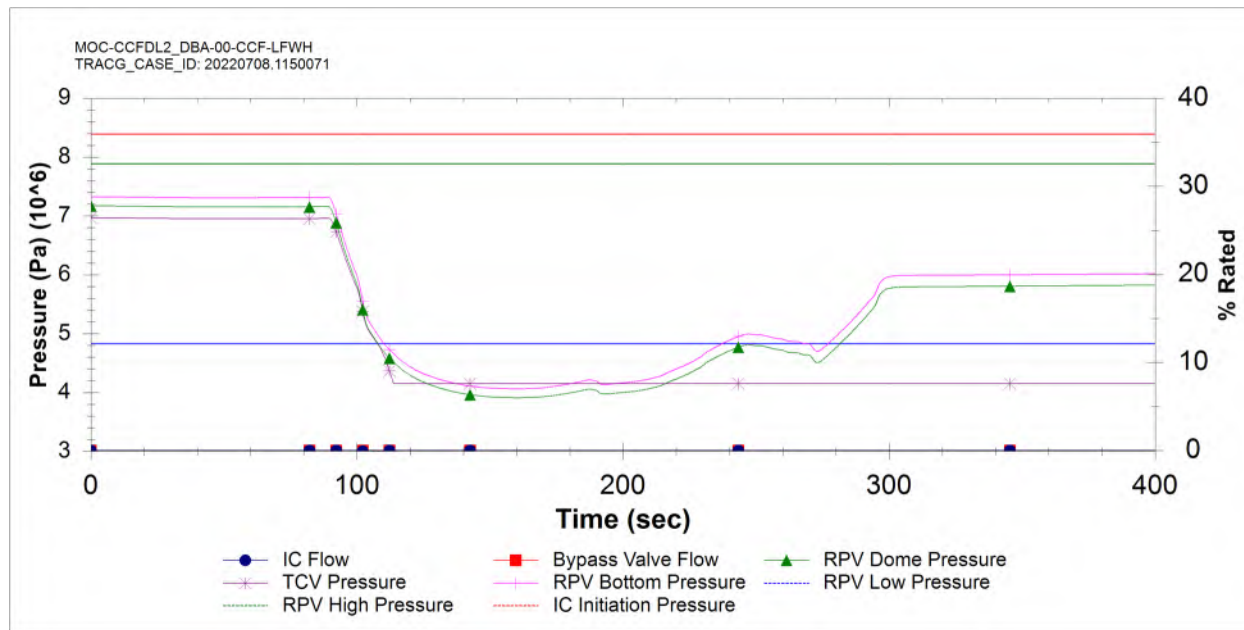


Figure 15.5-47: Loss of Feedwater Heating (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

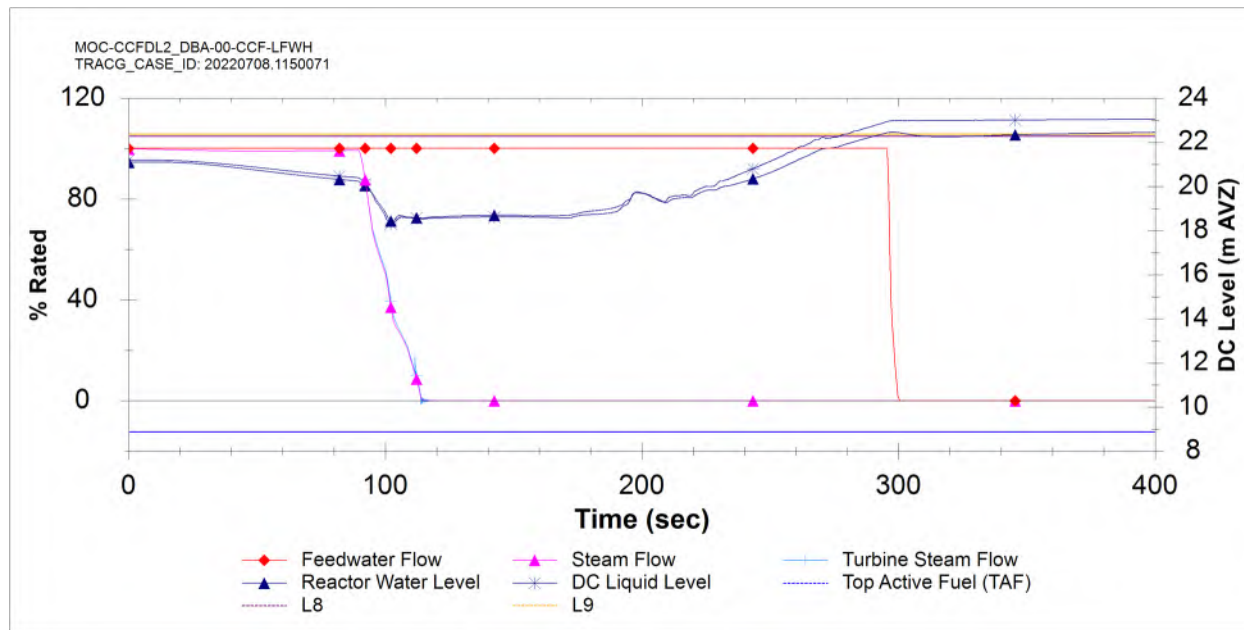
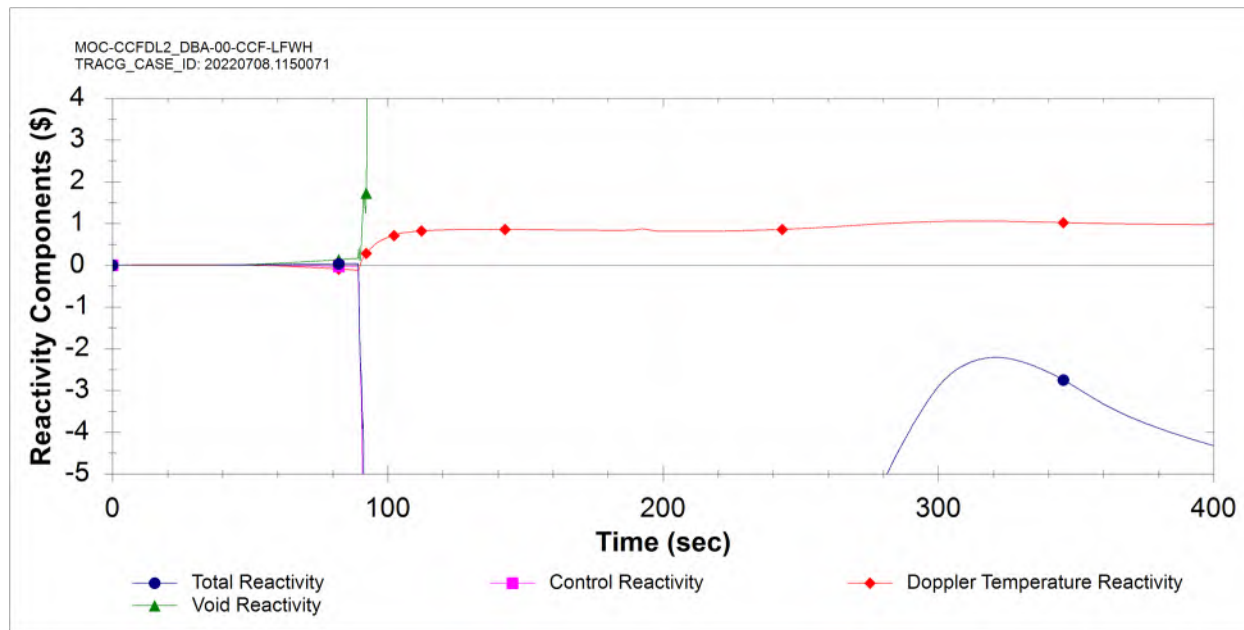


Figure 15.5-48: Loss of Feedwater Heating (DBA)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-49: Loss of Feedwater Heating (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

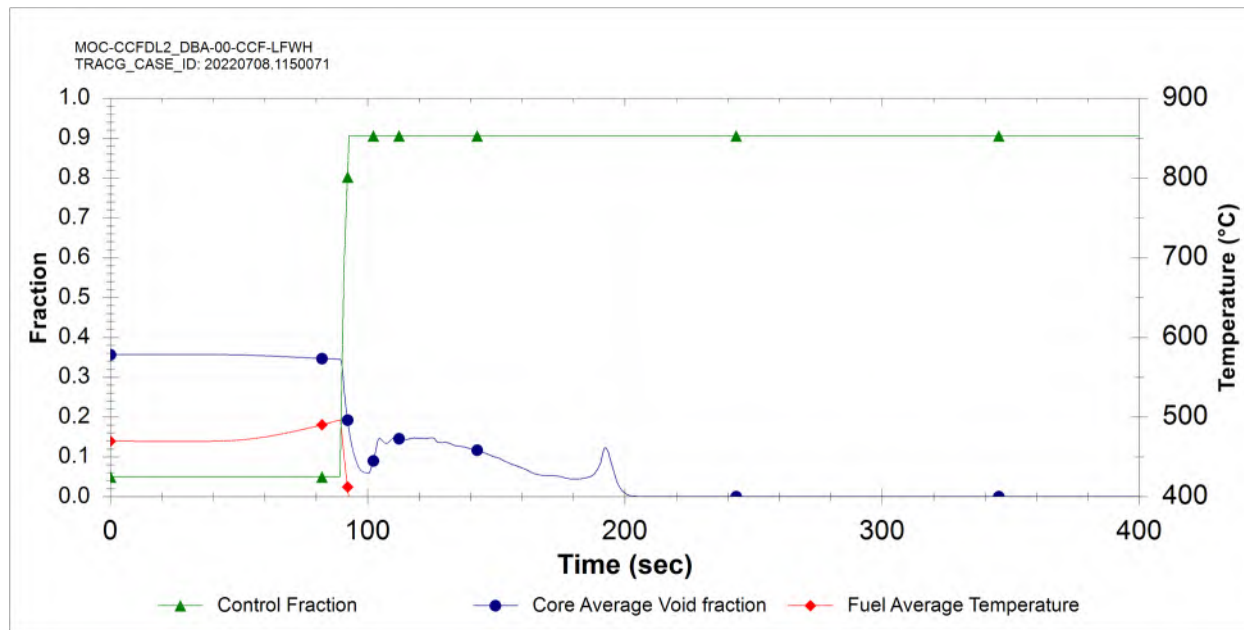
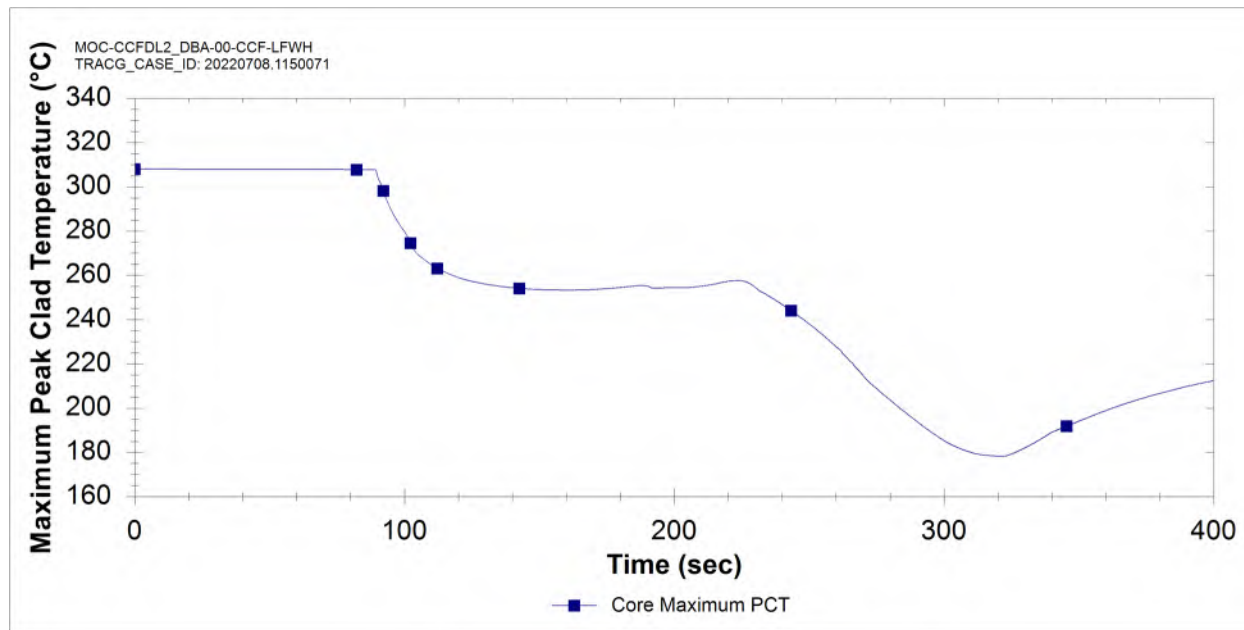


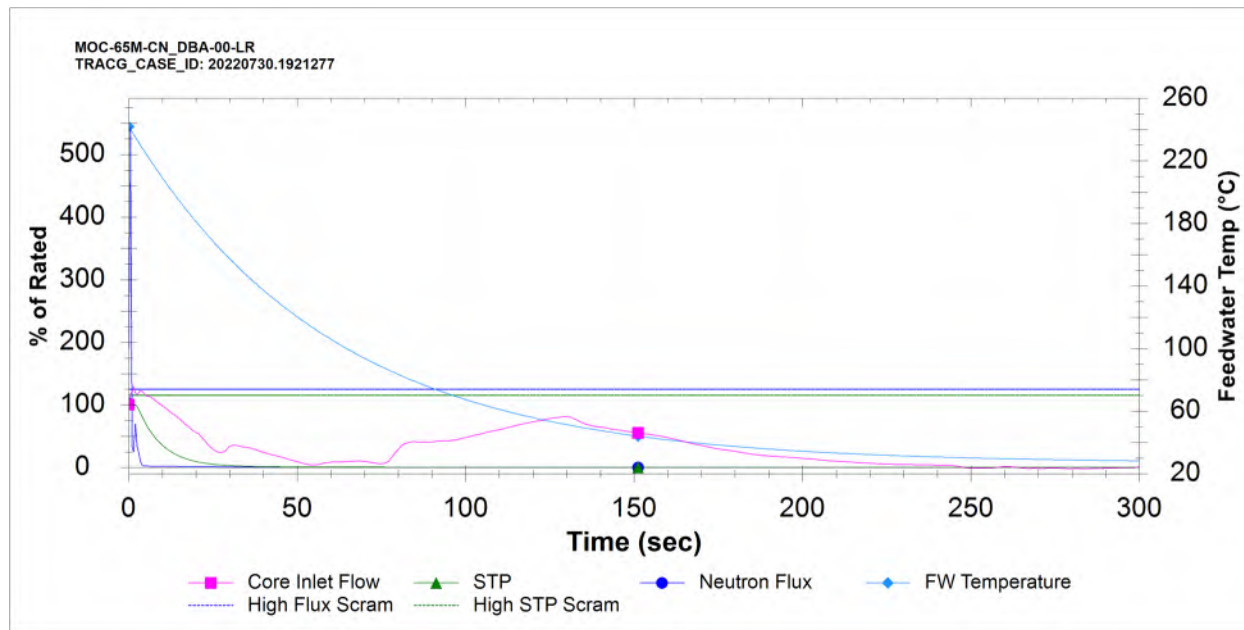
Figure 15.5-50: Loss of Feedwater Heating (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



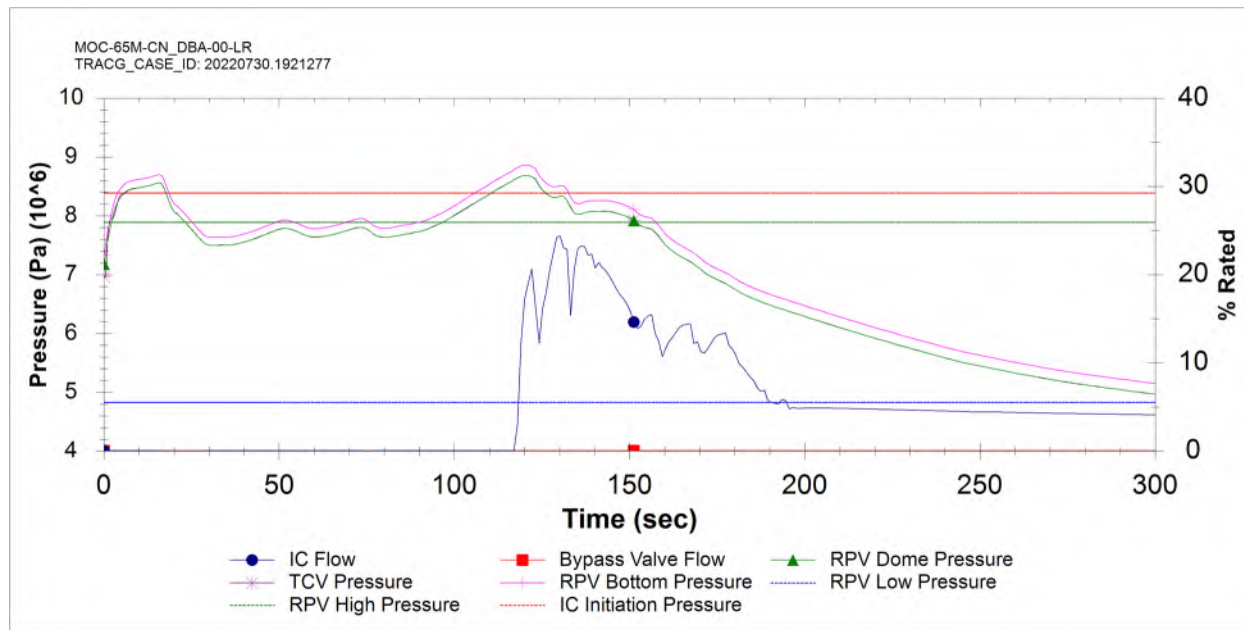
**Figure 15.5-51: Loss of Feedwater Heating (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-52: Generator Load Rejection (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-53: Generator Load Rejection (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

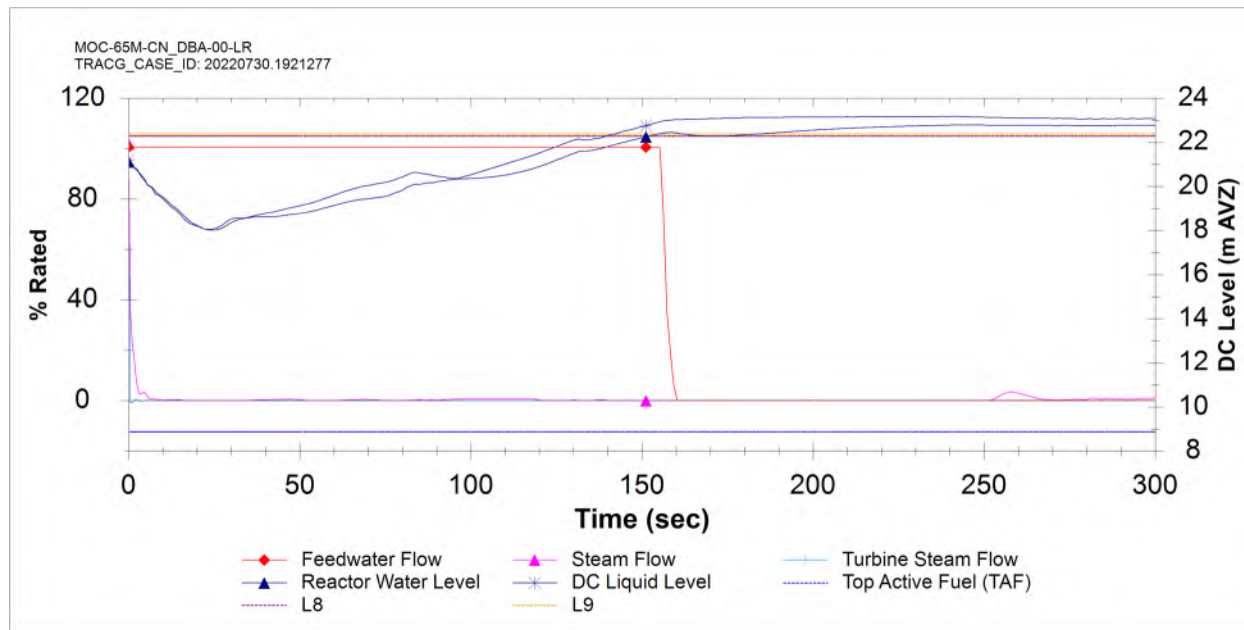
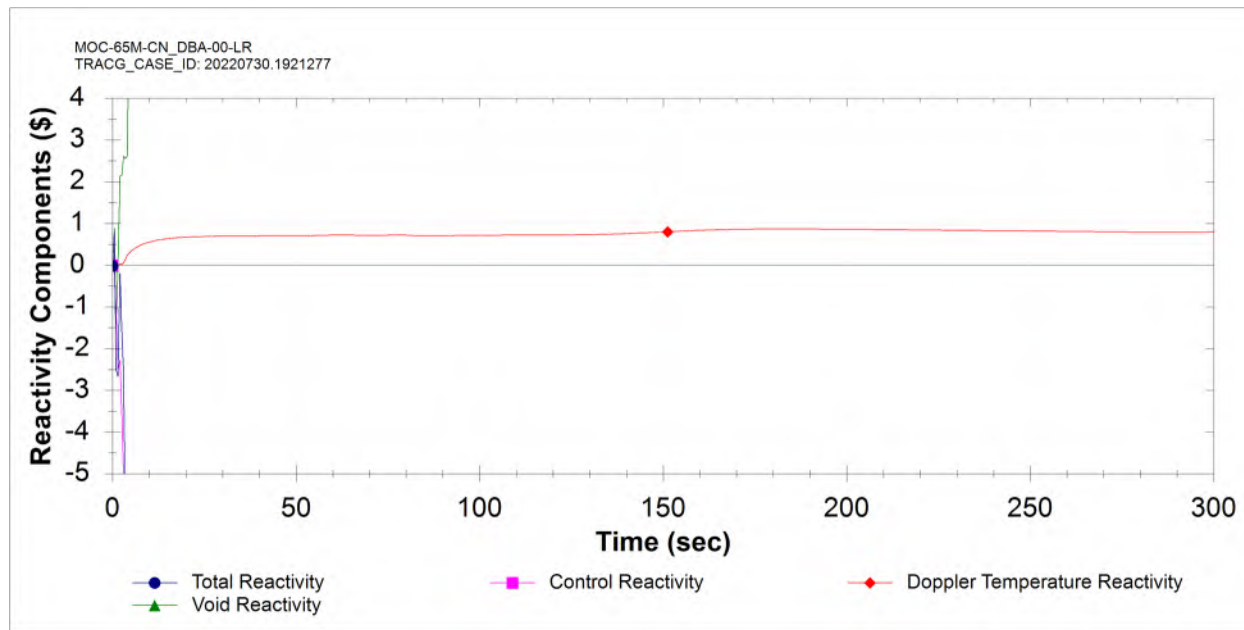


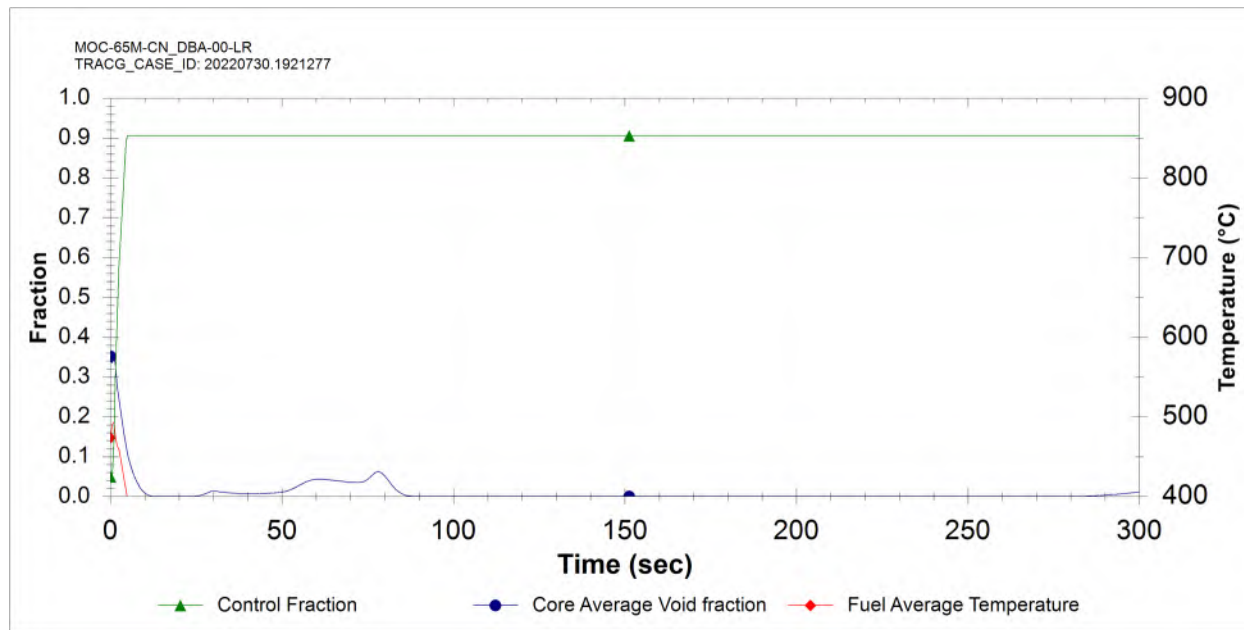
Figure 15.5-54: Generator Load Rejection (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-55: Generator Load Rejection (DBA)**

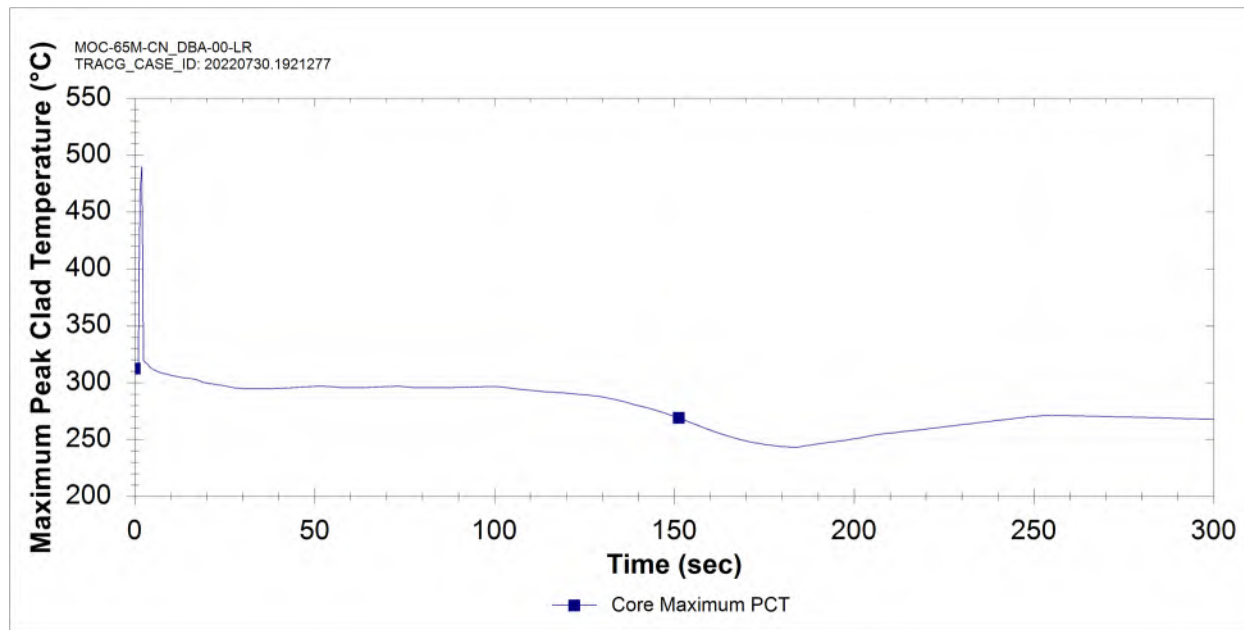
NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-56: Generator Load Rejection (DBA)**



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-57: Generator Load Rejection (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

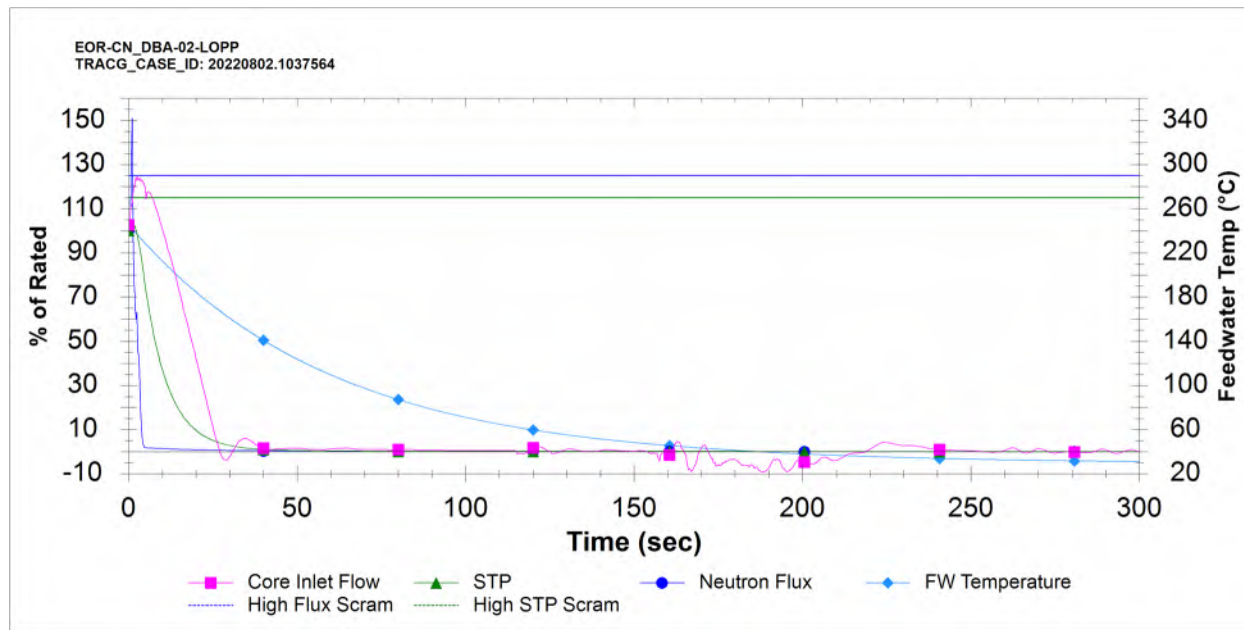


Figure 15.5-58: Loss of Preferred Power (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

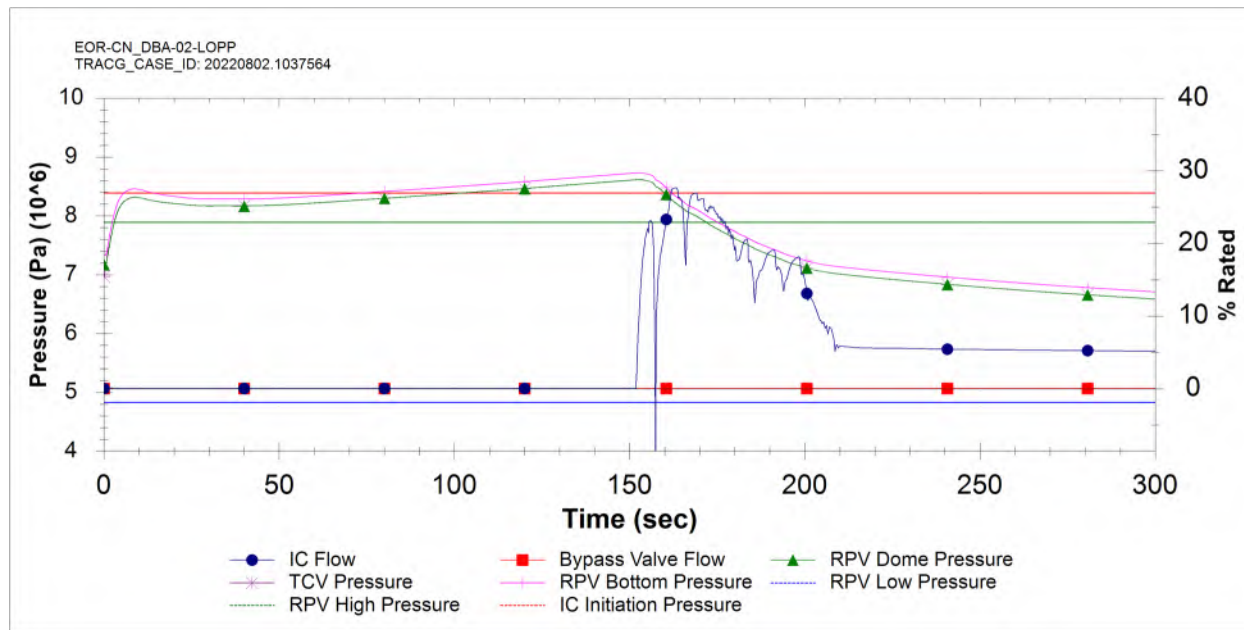
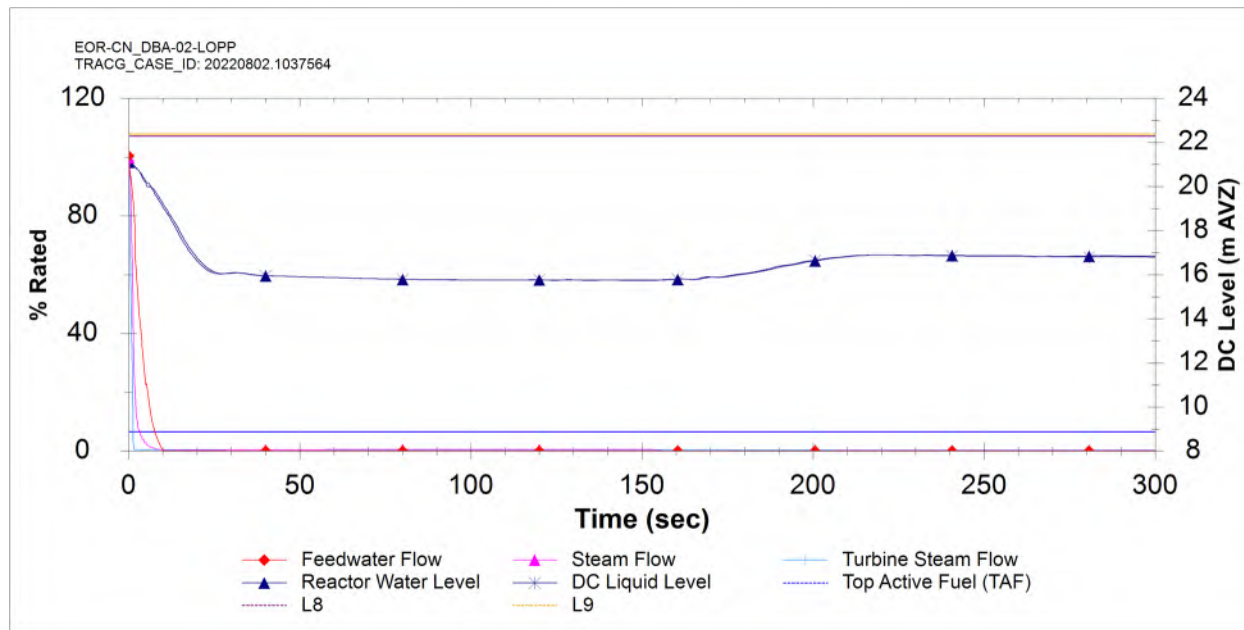


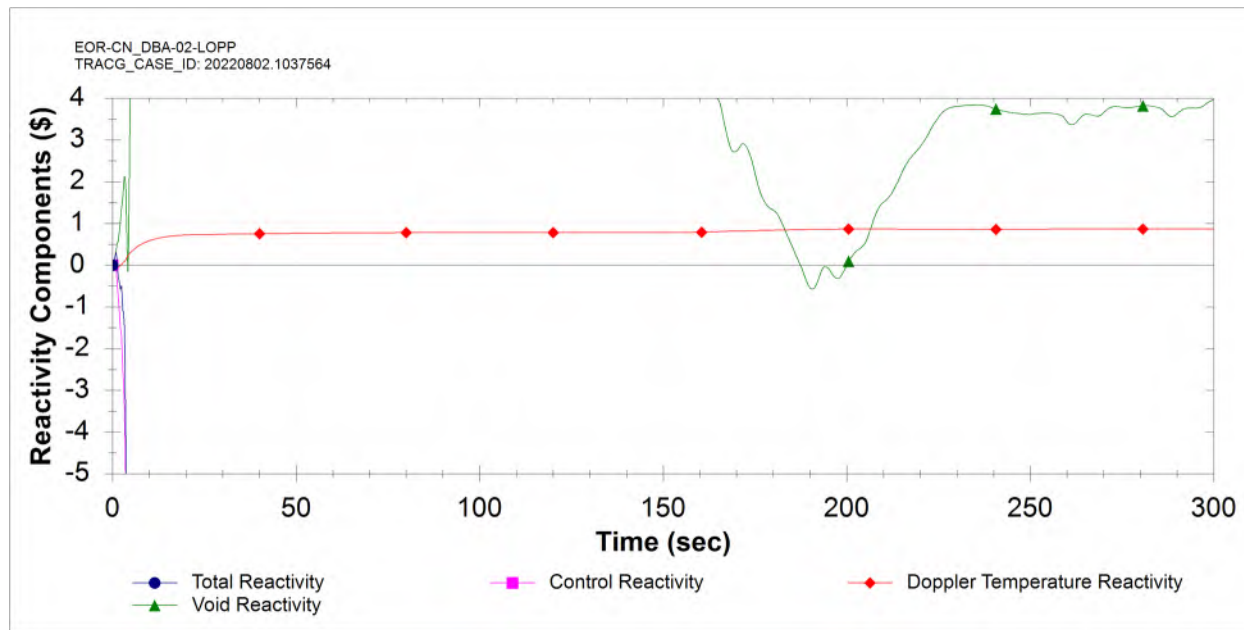
Figure 15.5-59: Loss of Preferred Power (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-60: Loss of Preferred Power (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-61: Loss of Preferred Power (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

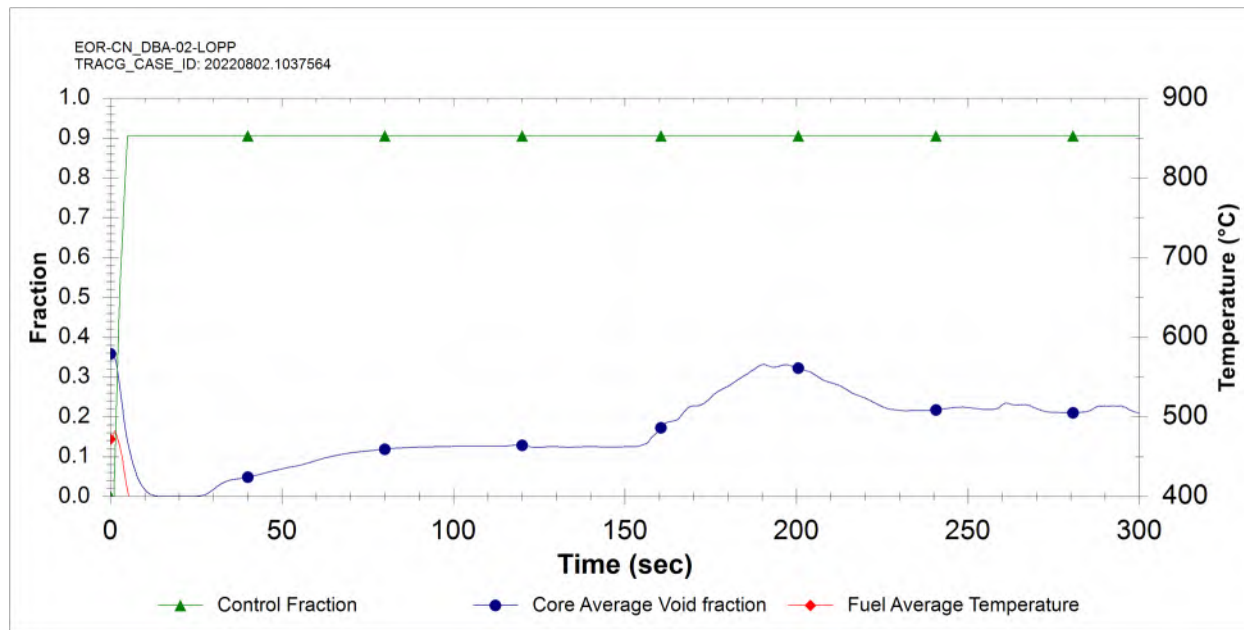
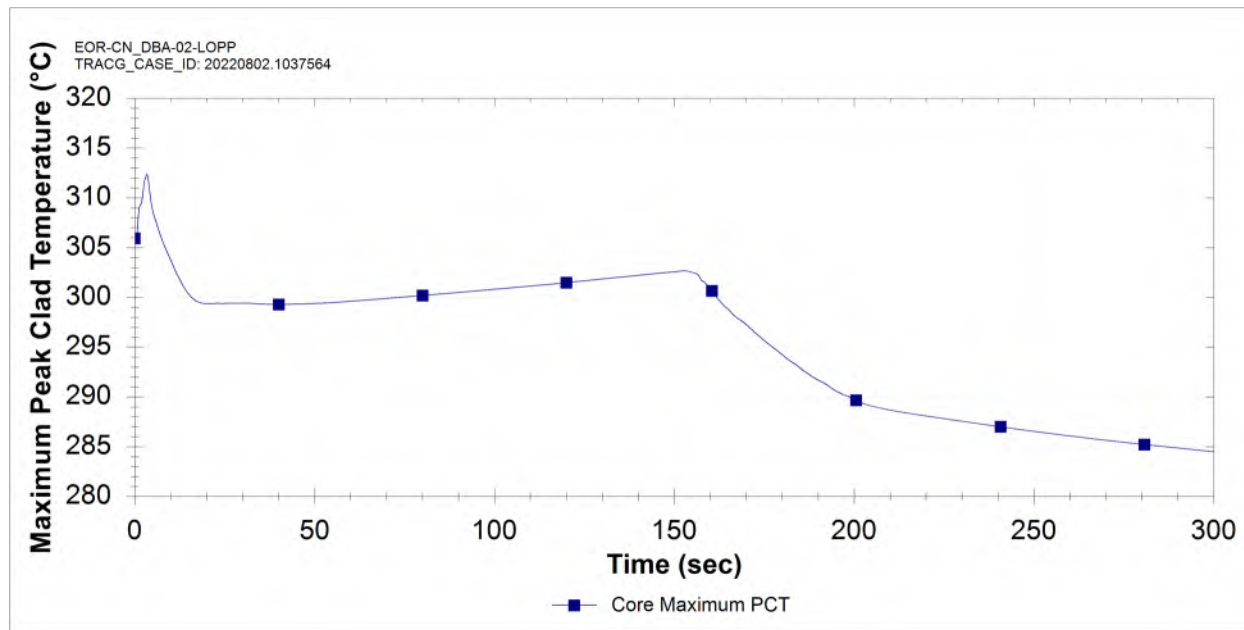


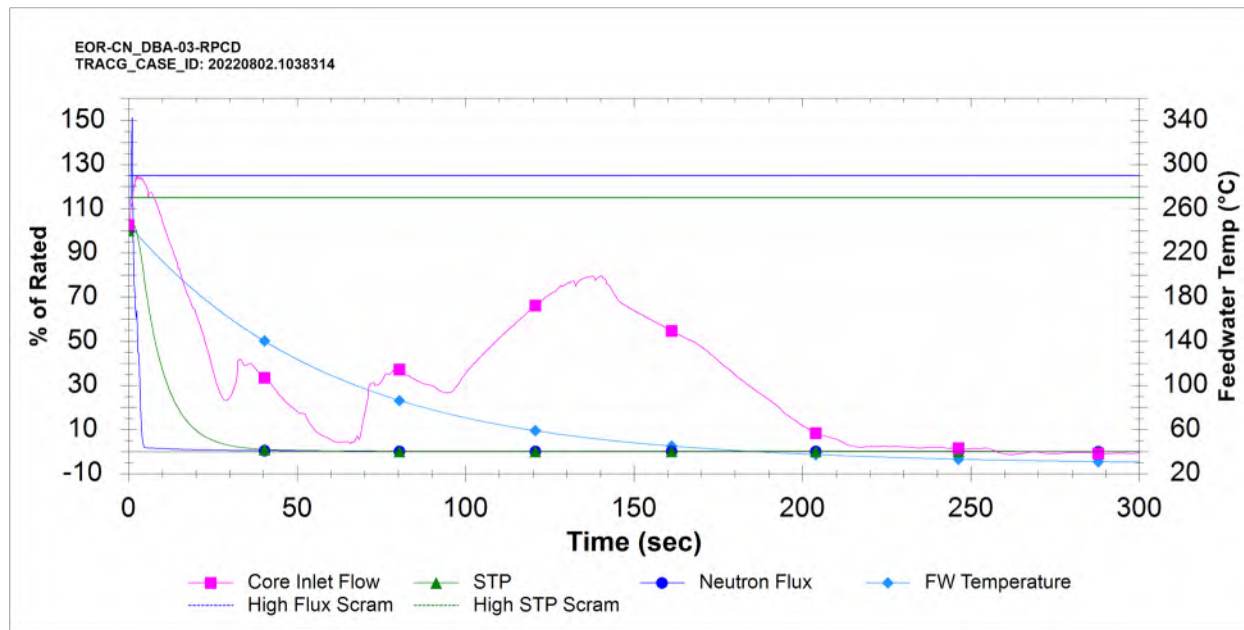
Figure 15.5-62: Loss of Preferred Power (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-63: Loss of Preferred Power (DBA)**

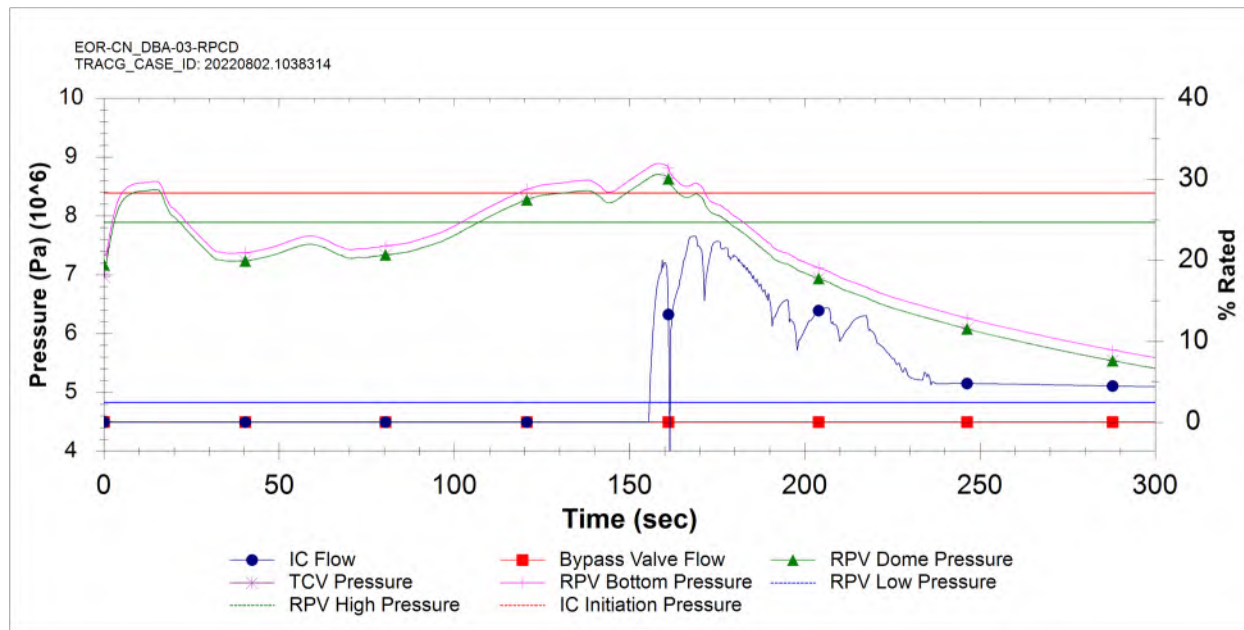
NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-64: RPV Pressure Control Downscale (DBA)**



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-65: RPV Pressure Control Downscale (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

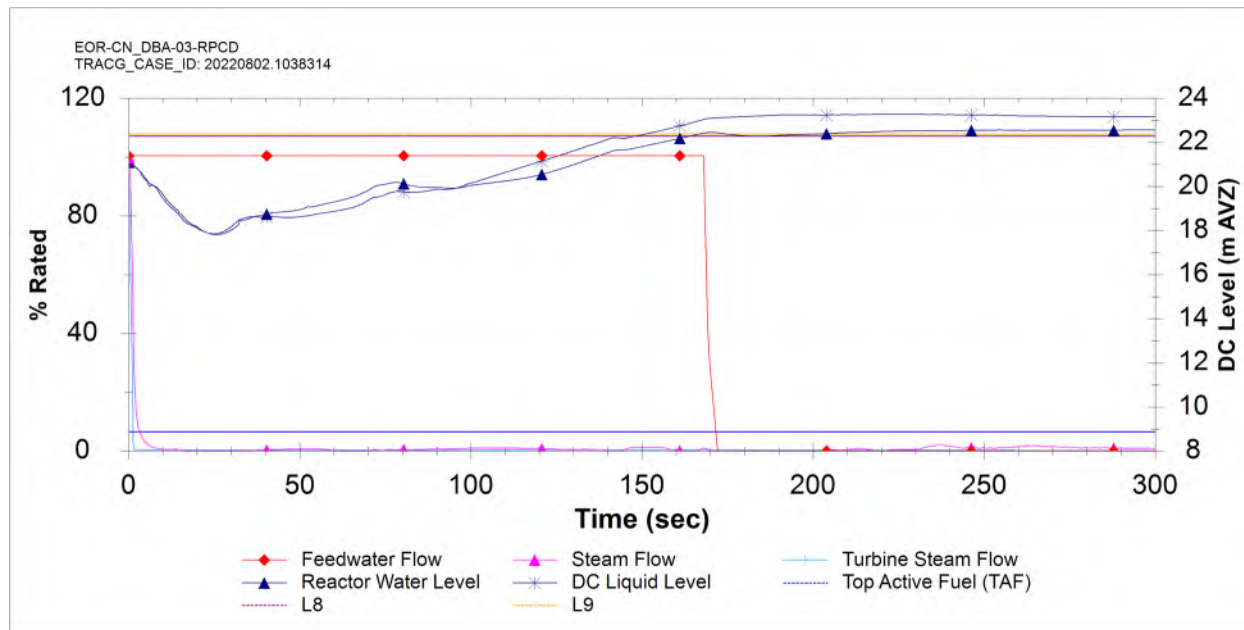
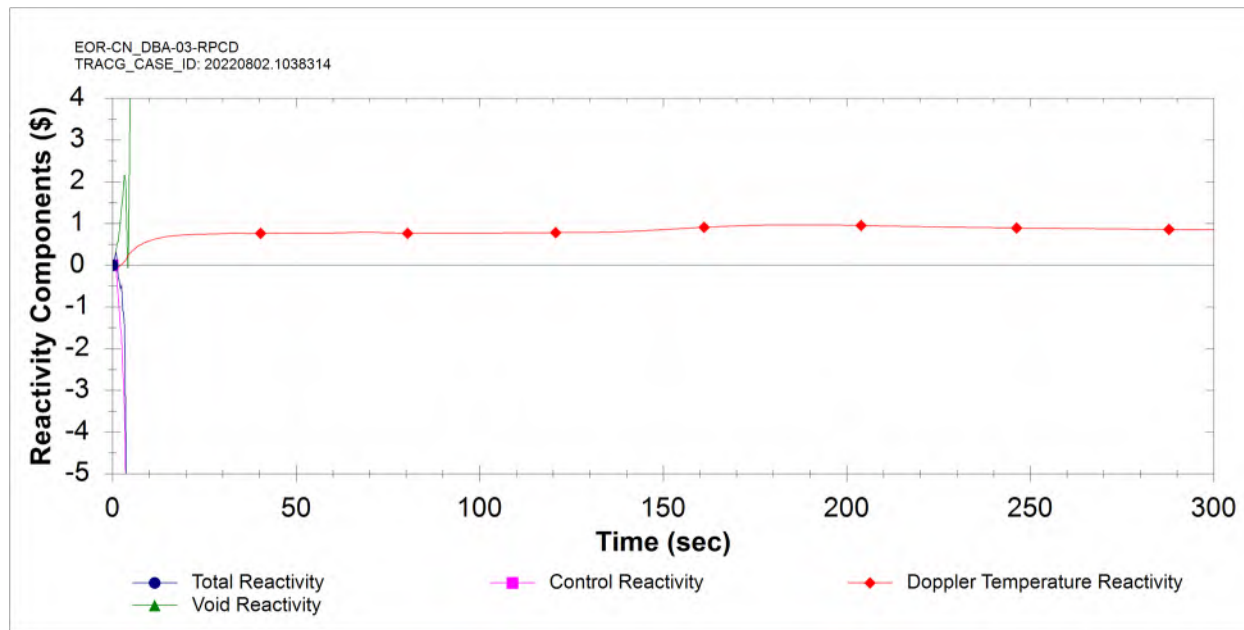


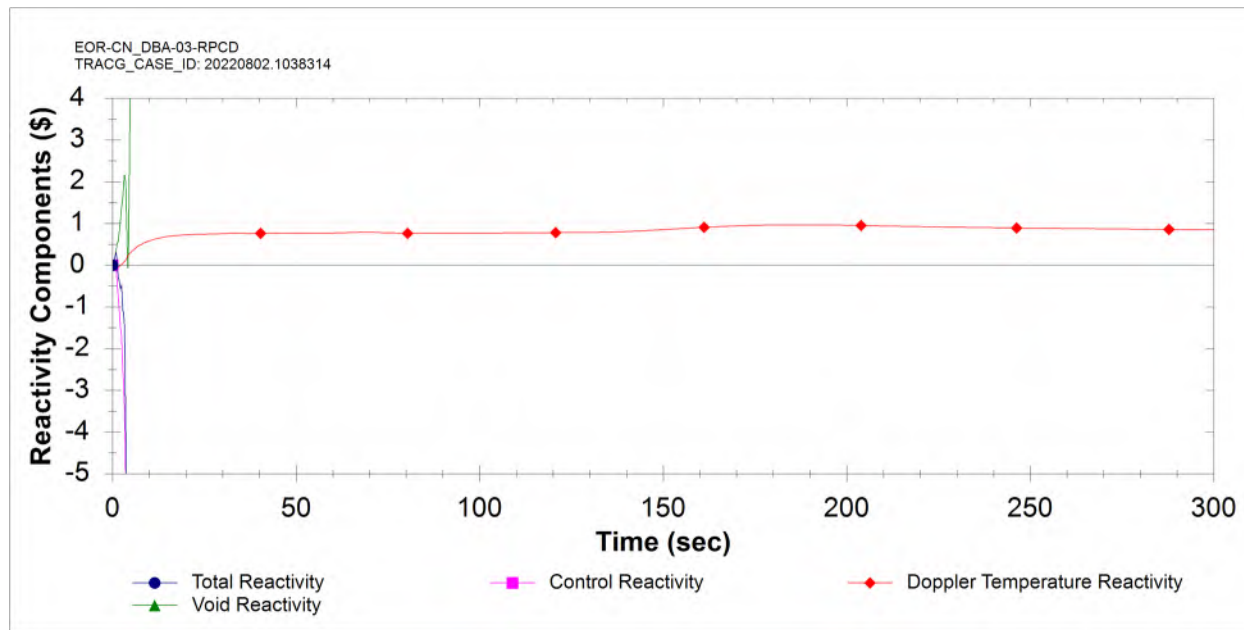
Figure 15.5-66: RPV Pressure Control Downscale (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



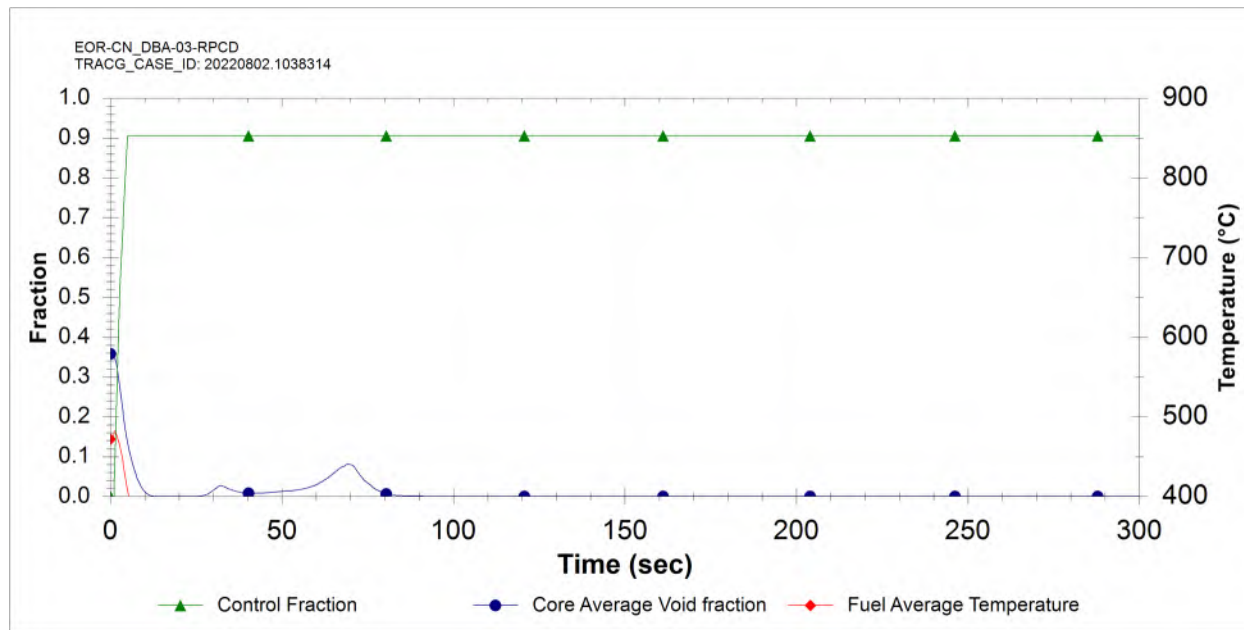
**Figure 15.5-67: RPV Pressure Control Downscale (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.6-68a: RPV Pressure Control Downscale (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-68b: RPV Pressure Control Downscale (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

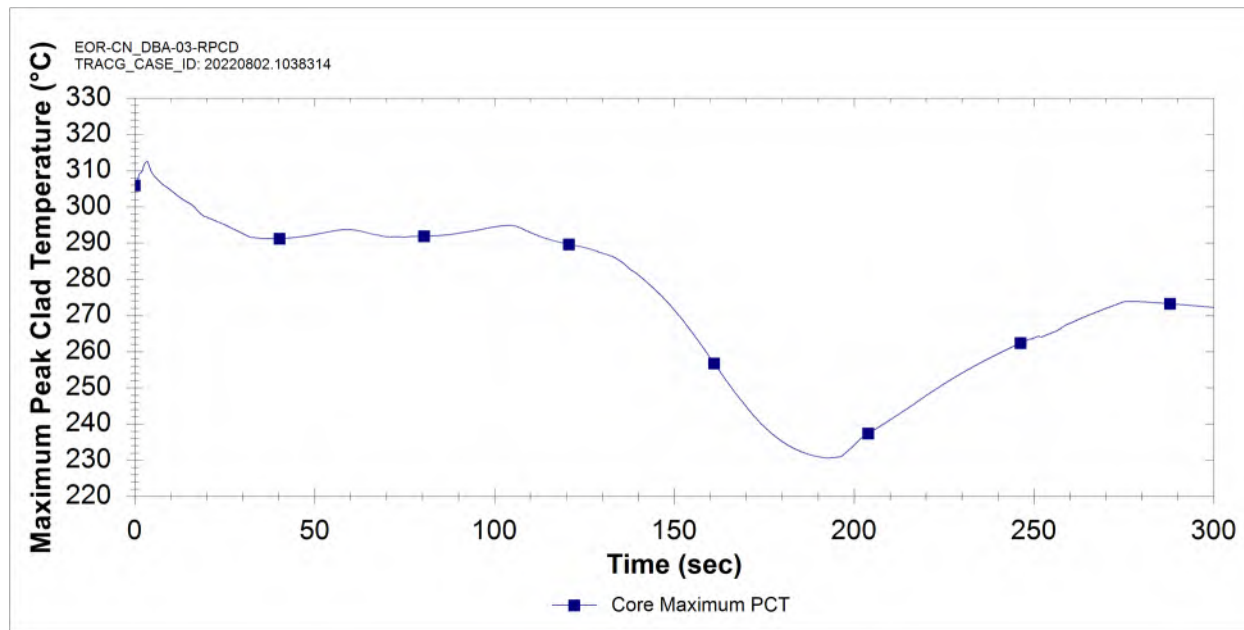
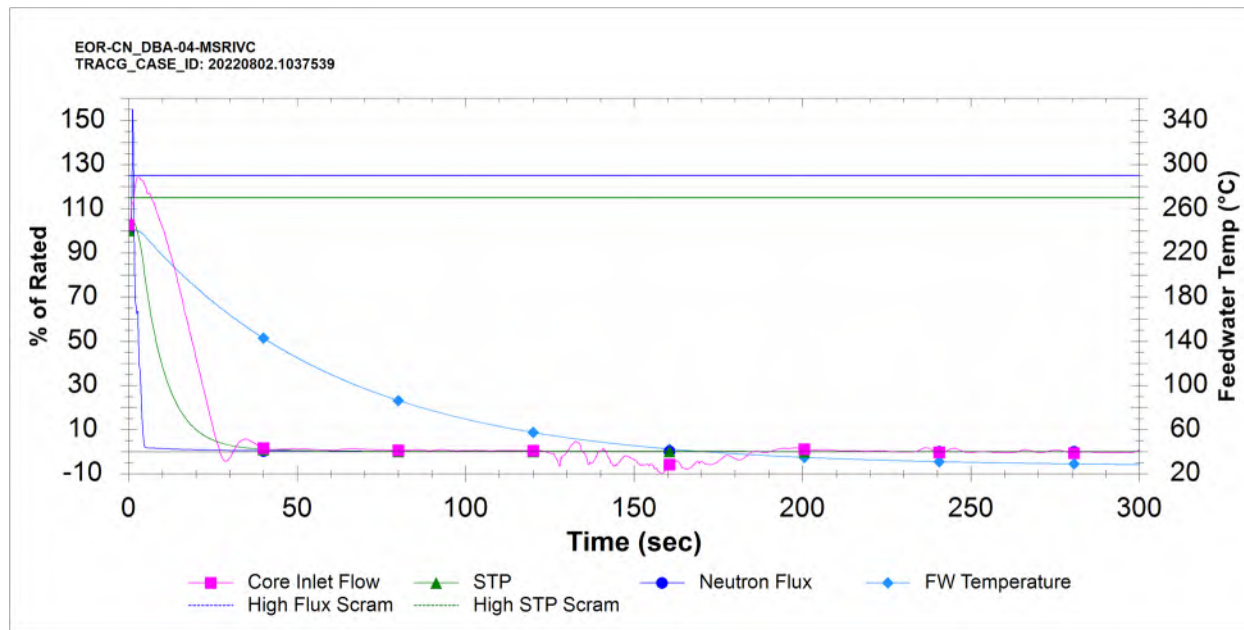


Figure 15.5-69: RPV Pressure Control Downscale (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-70: Closure of All MSRIVs and FW Isolation Valves (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

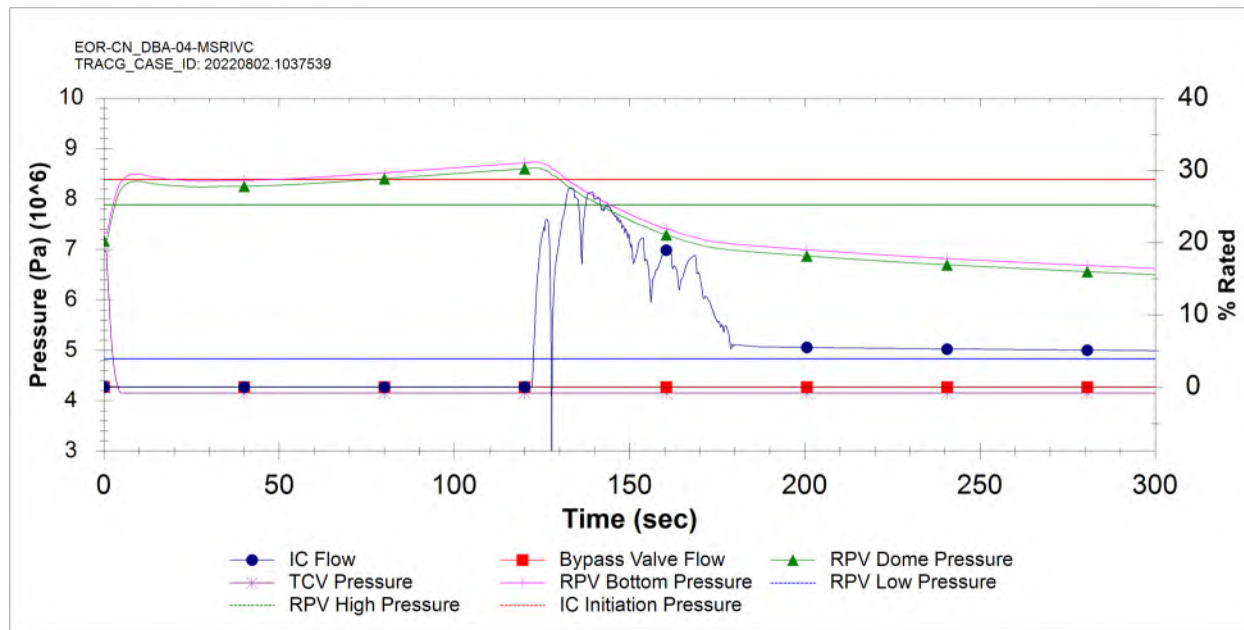


Figure 15.5-71: Closure of All MSRIVs and FW Isolation Valves (DBA)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

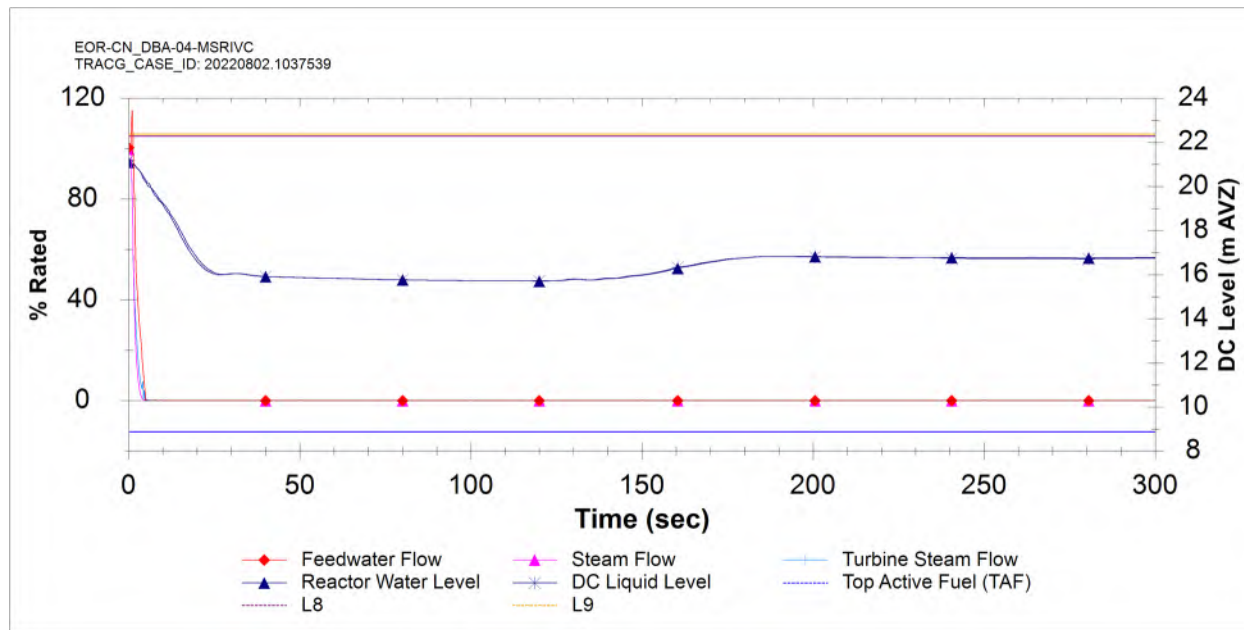
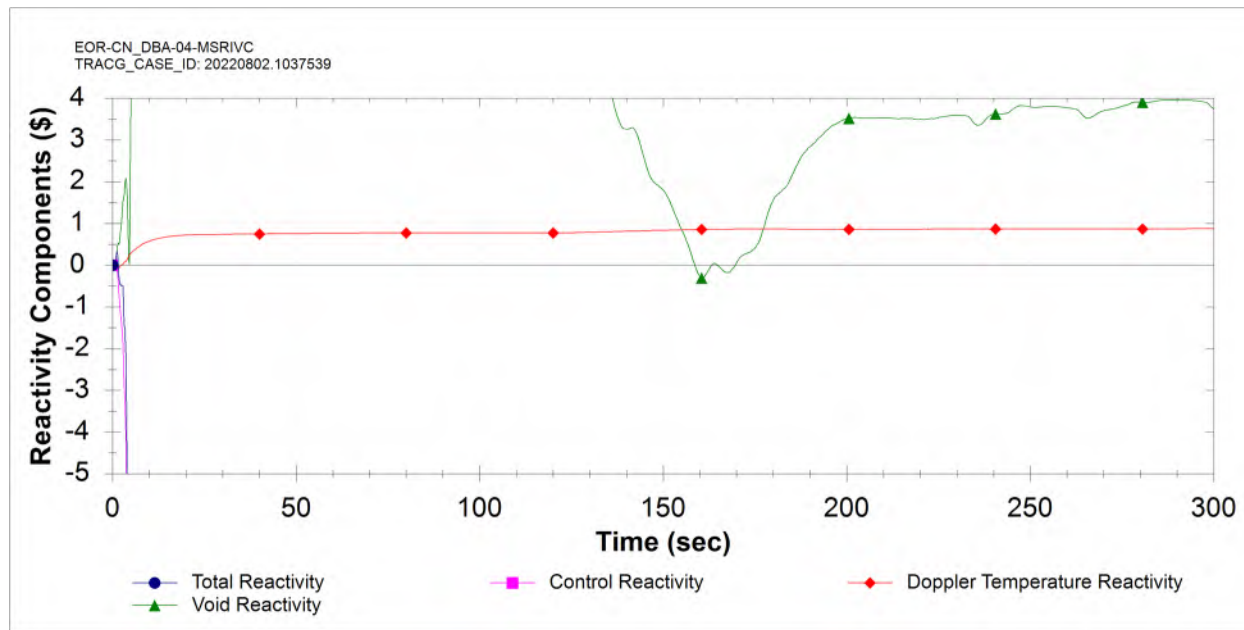


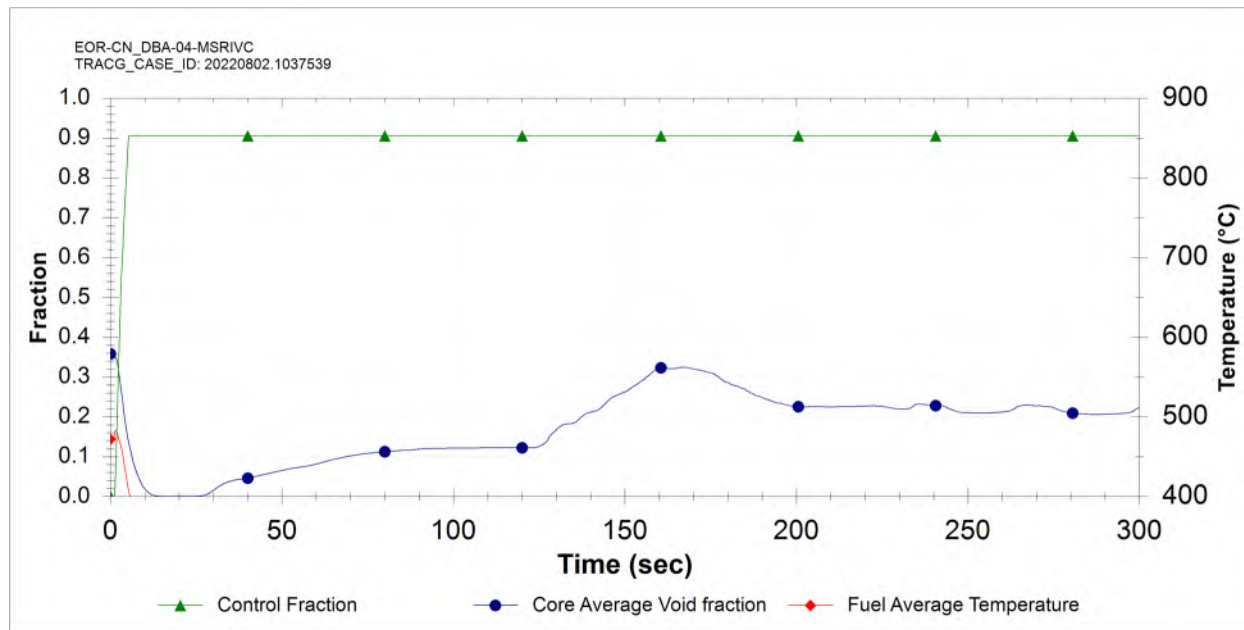
Figure 15.5-72: Closure of All MSRIVs and FW Isolation Valves (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-73: Closure of All MSRIVs and FW Isolation Valves (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-74: Closure of All MSRIVs and FW Isolation Valves (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

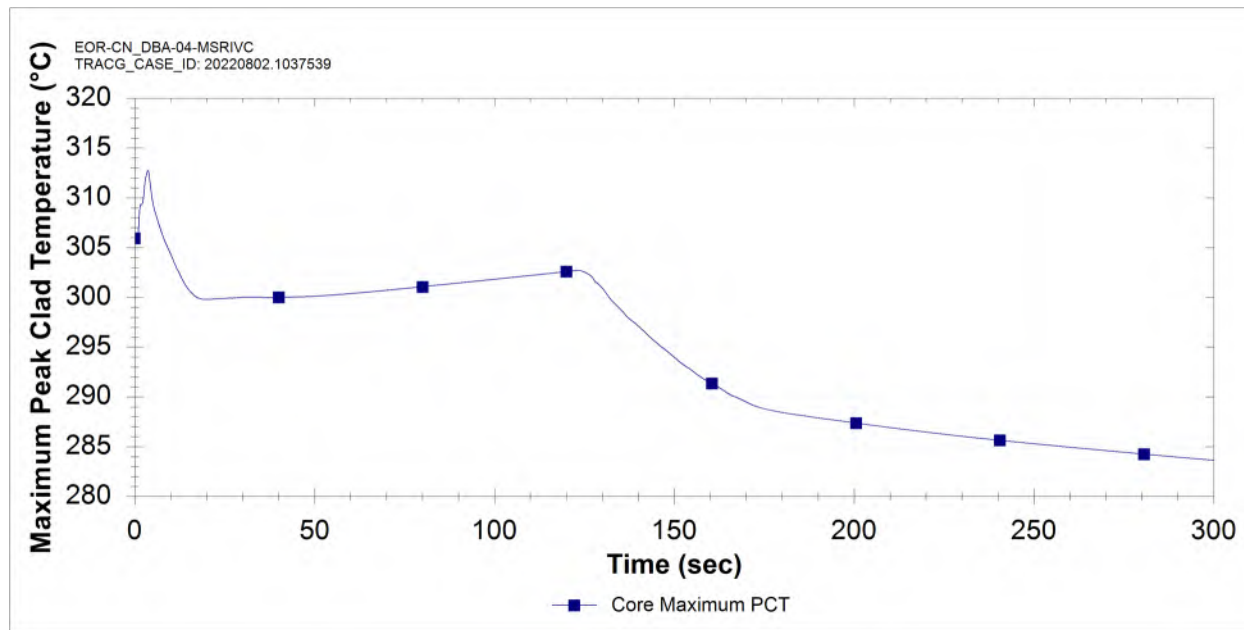


Figure 15.5-75: Closure of All MSRIVs and FW Isolation Valves (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

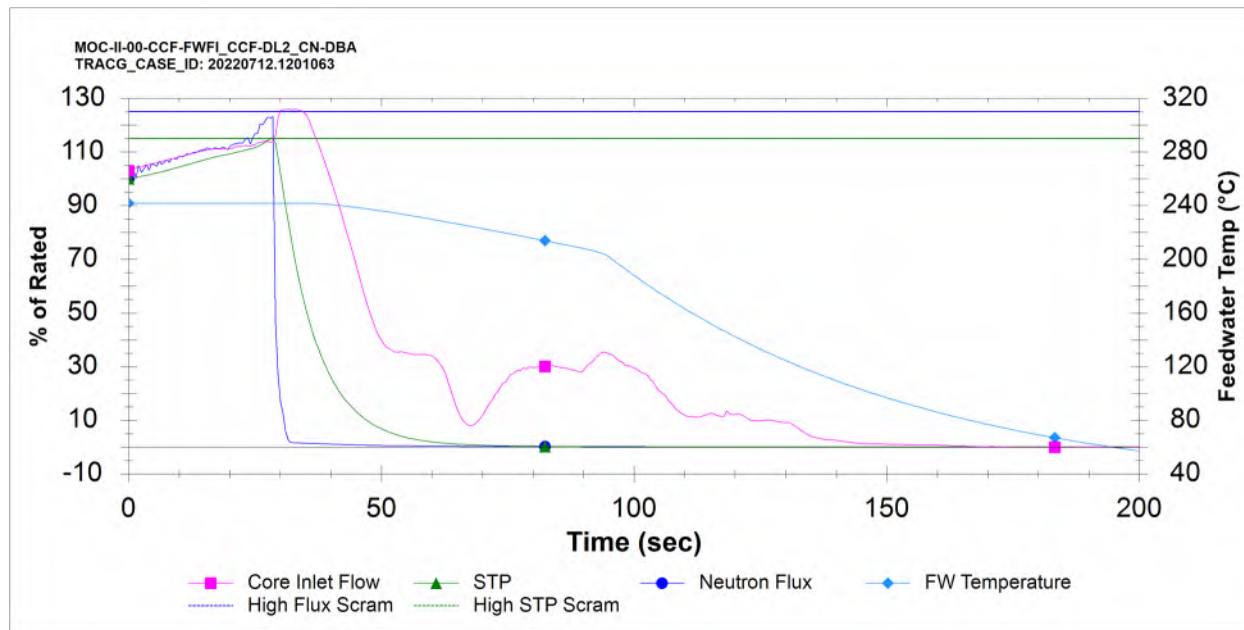
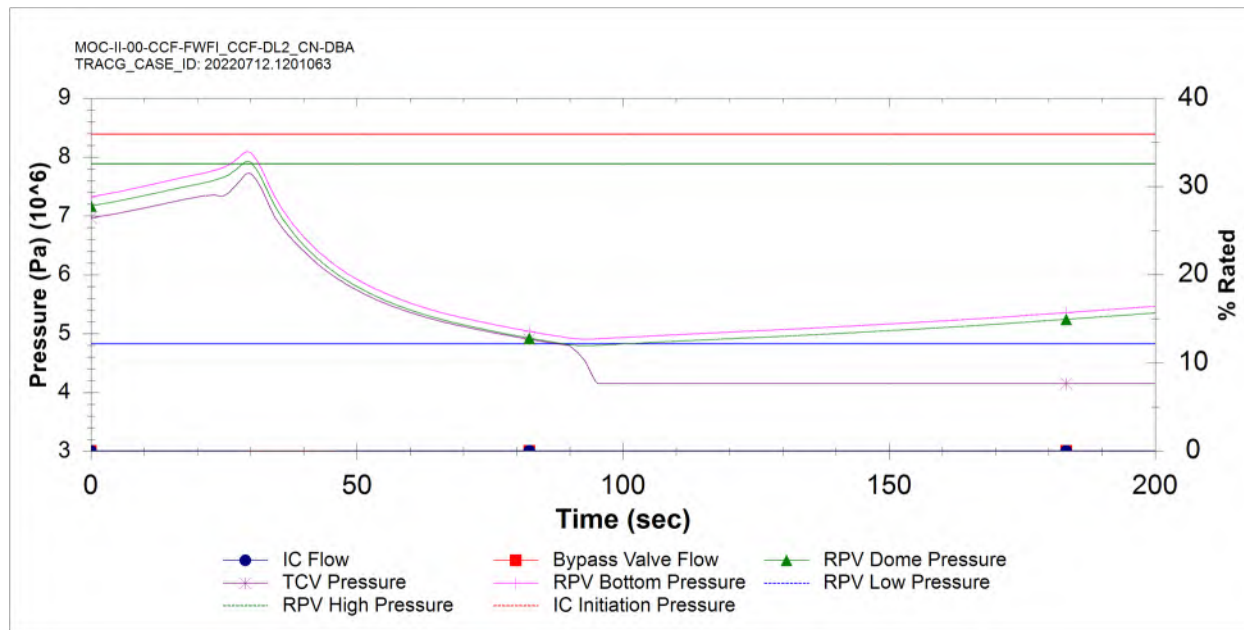


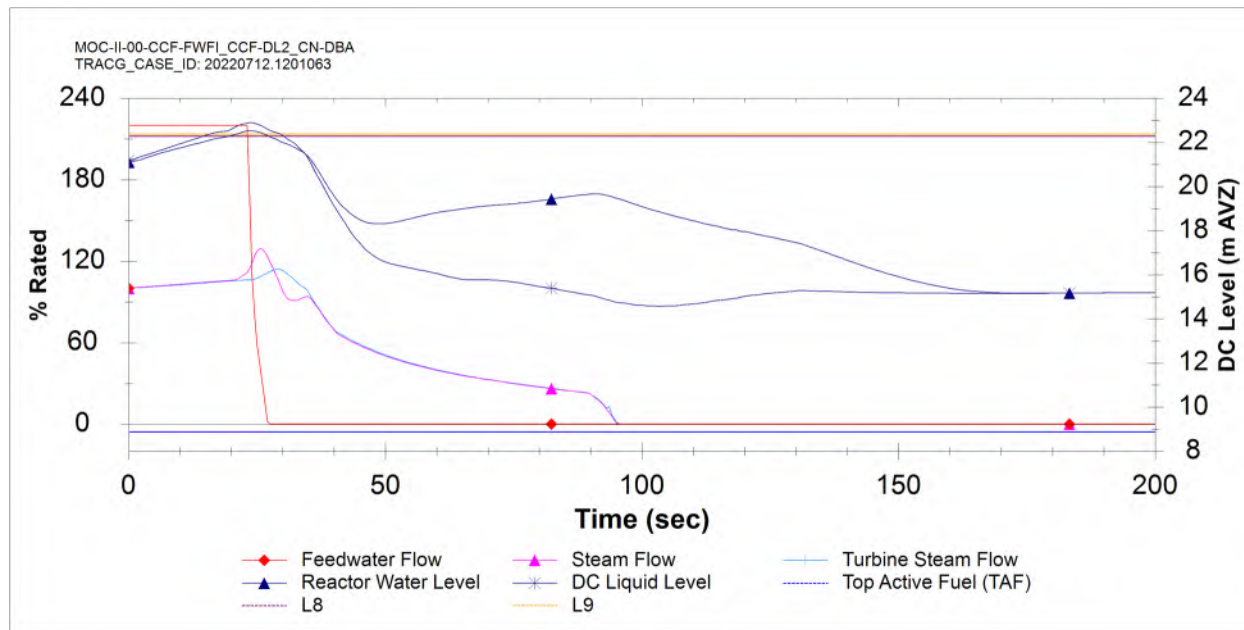
Figure 15.5-76: Feedwater Flow Increase (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



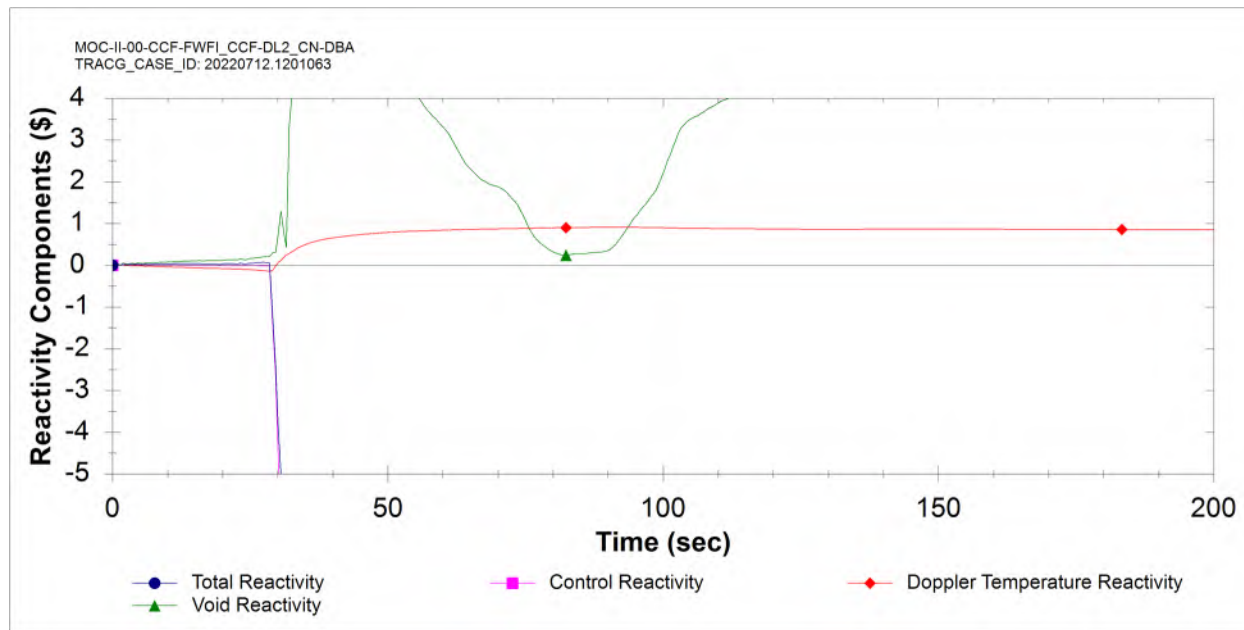
**Figure 15.5-77: Feedwater Flow Increase (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-78: Feedwater Flow Increase (DBA)**

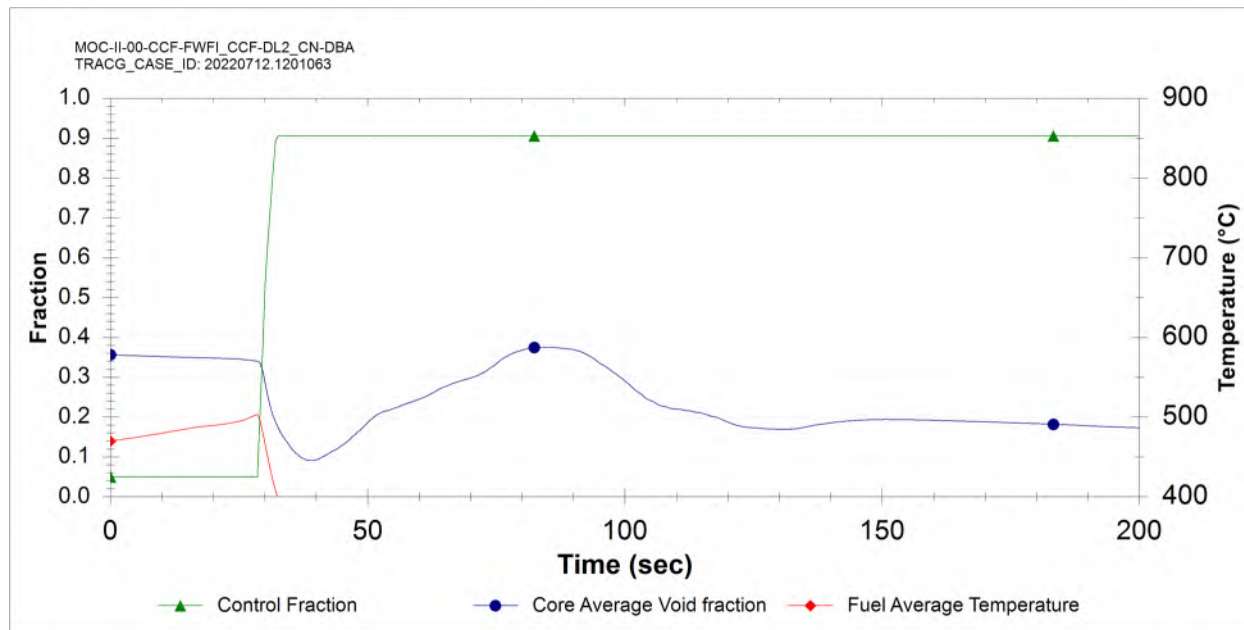
NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-79: Feedwater Flow Increase (DBA)**

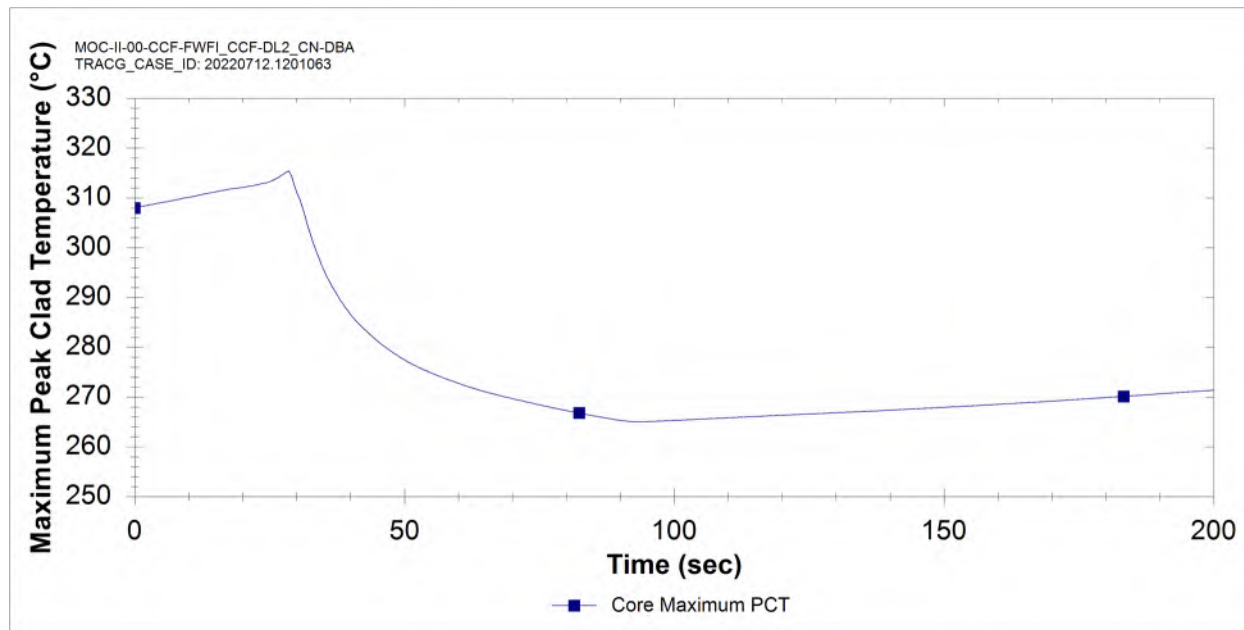


NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



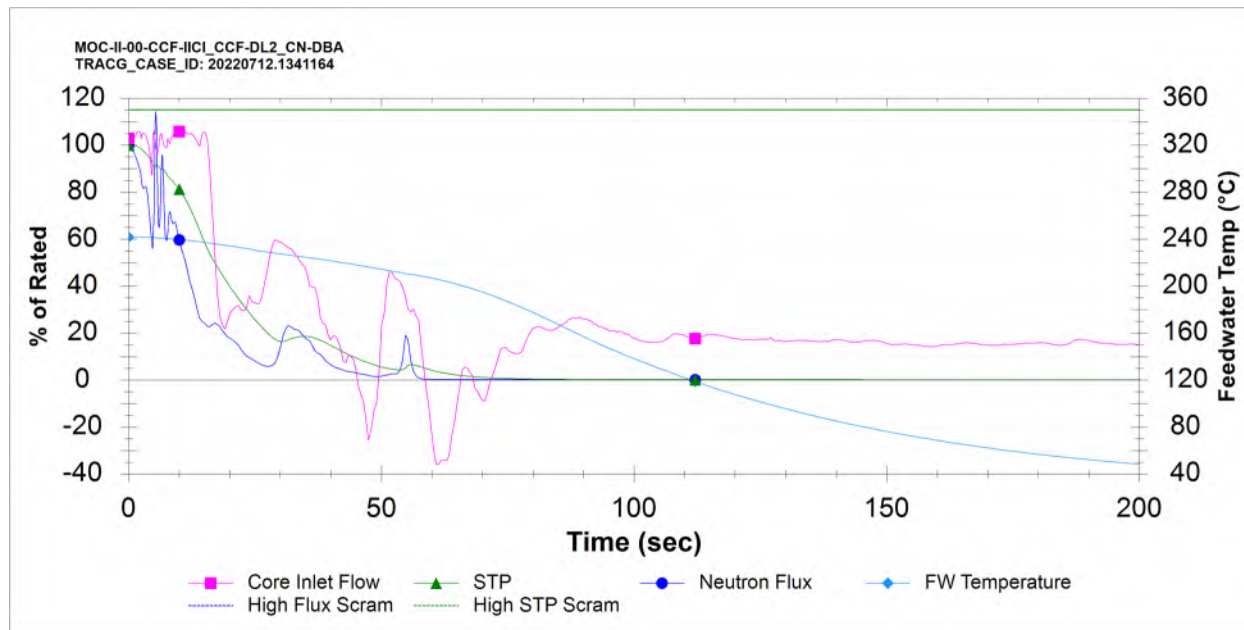
**Figure 15.5-80: Feedwater Flow Increase (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-81: Feedwater Flow Increase (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-82: Inadvertent Isolation Condenser Initiation - All Trains (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

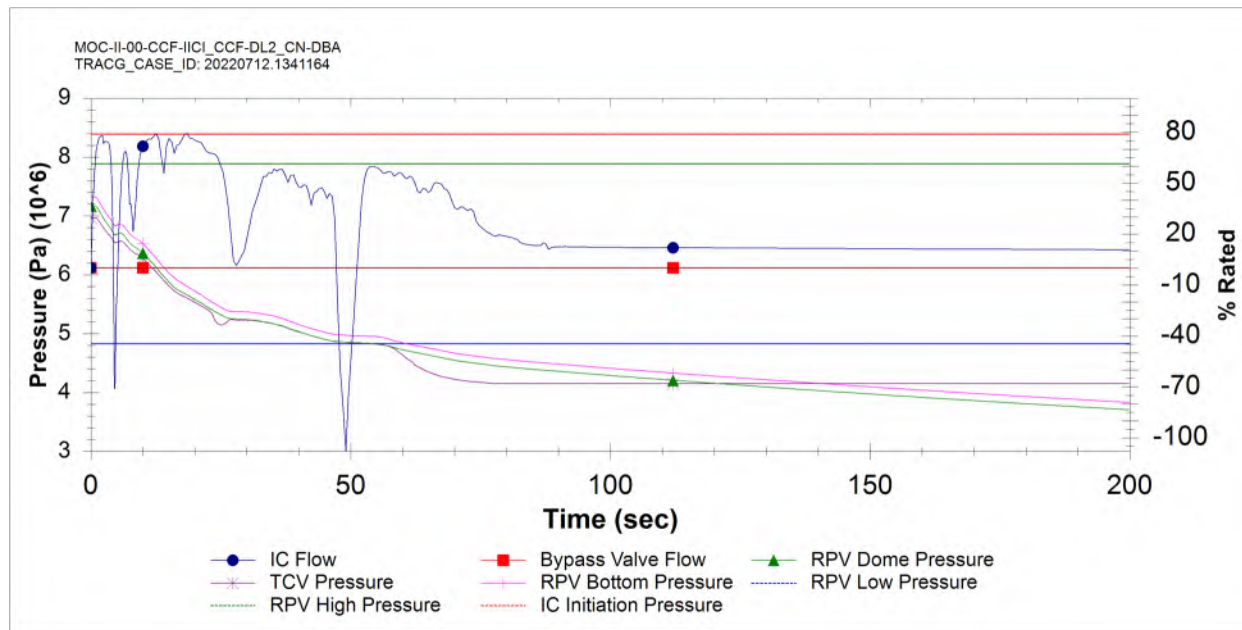


Figure 15.5-83: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

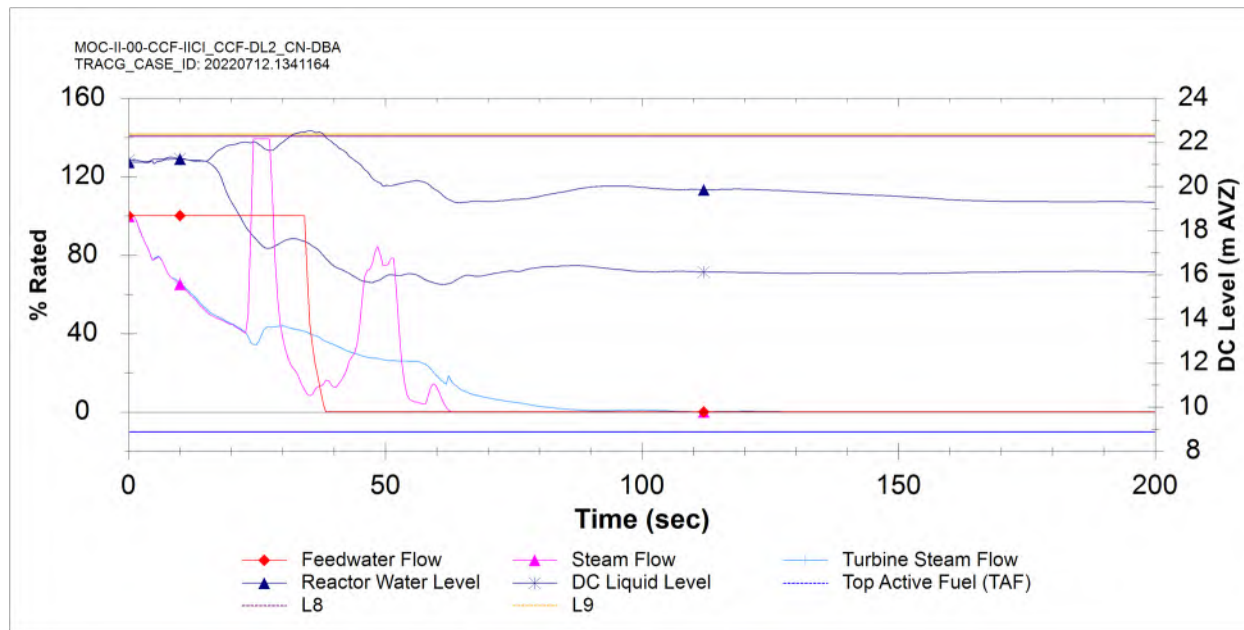
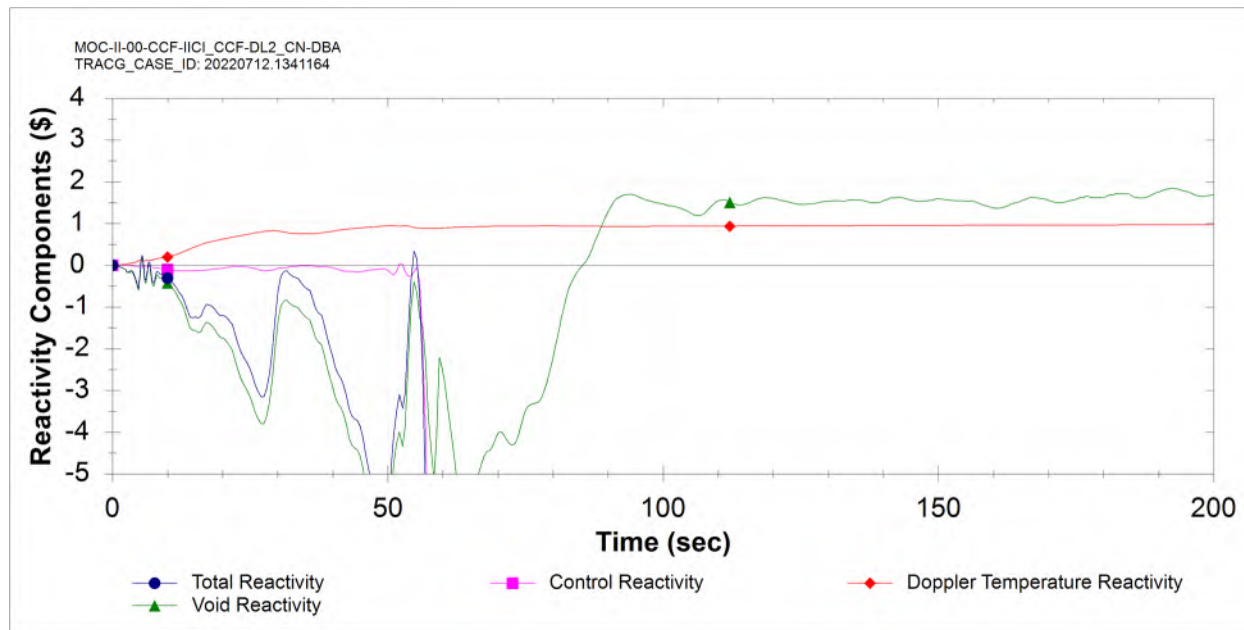


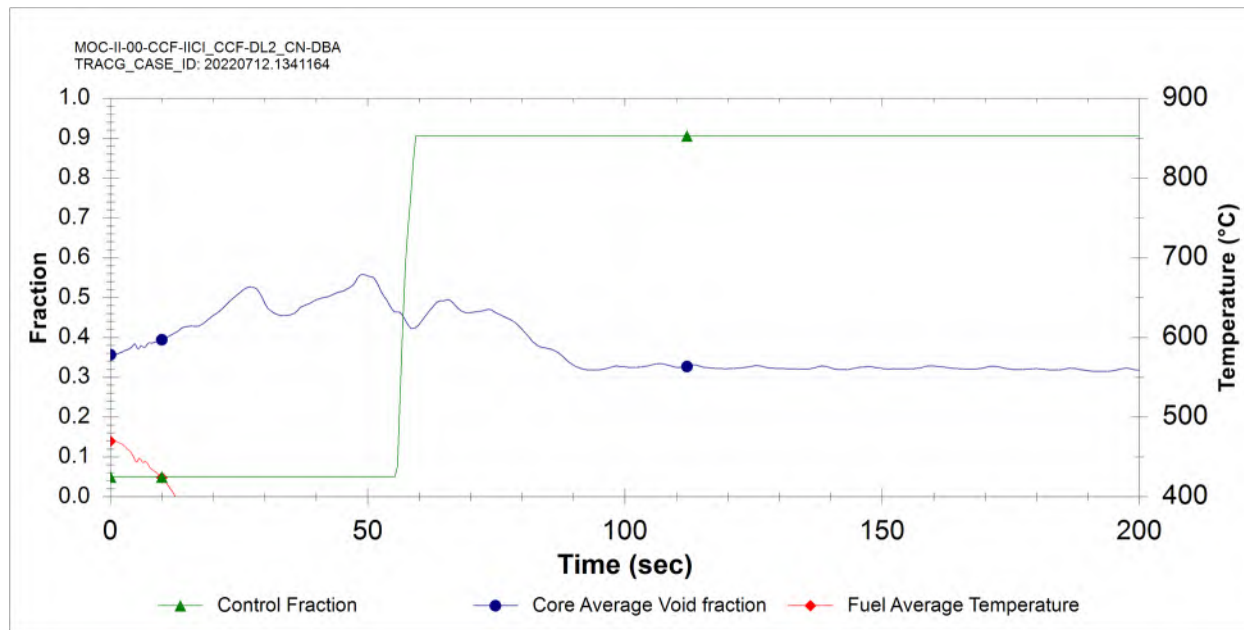
Figure 15.5-84: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-85: Inadvertent Isolation Condenser Initiation - All Trains (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-86: Inadvertent Isolation Condenser Initiation - All Trains (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

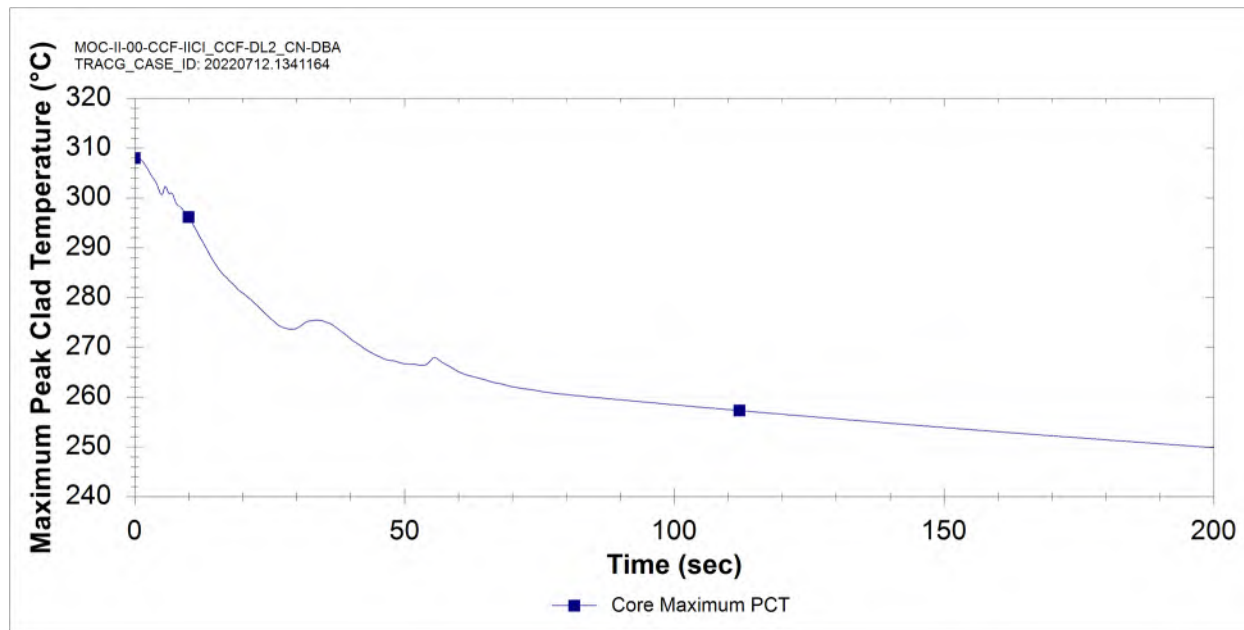
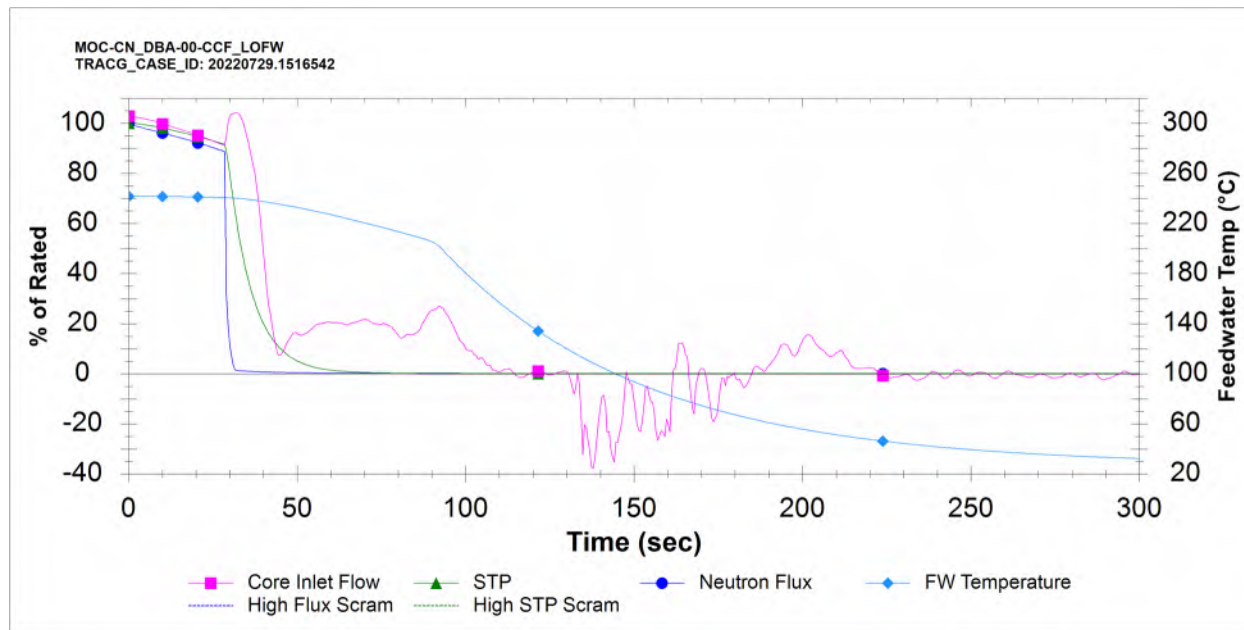


Figure 15.5-87: Inadvertent Isolation Condenser Initiation - All Trains (DBA)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-88: Loss of Feedwater Flow (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

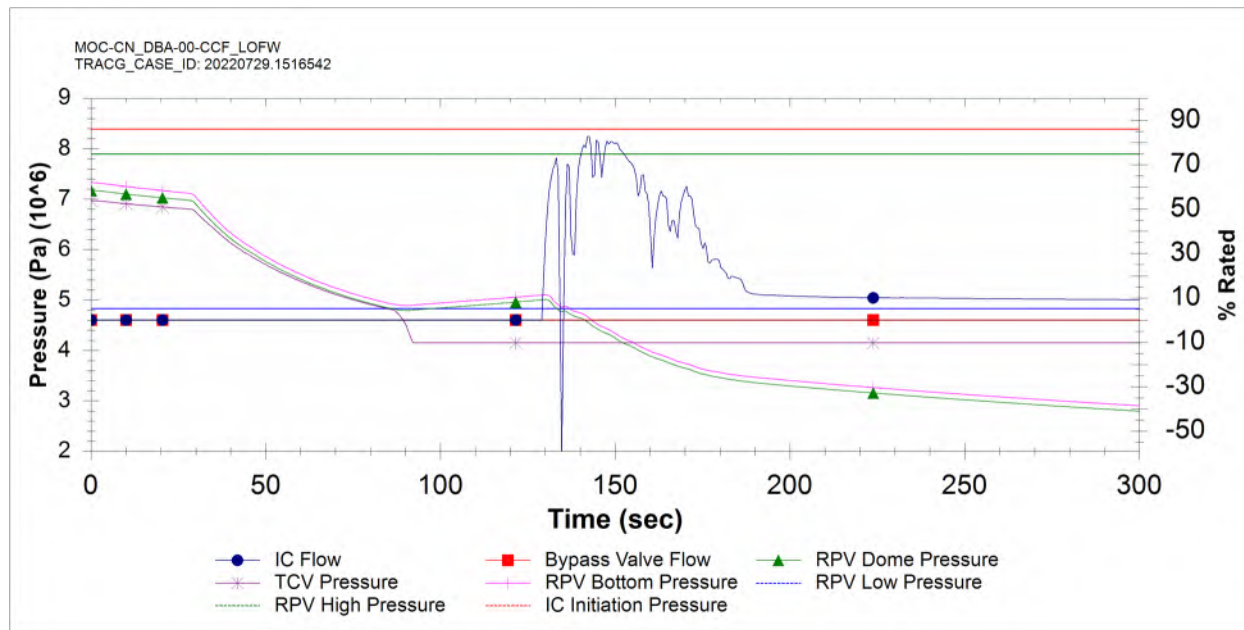
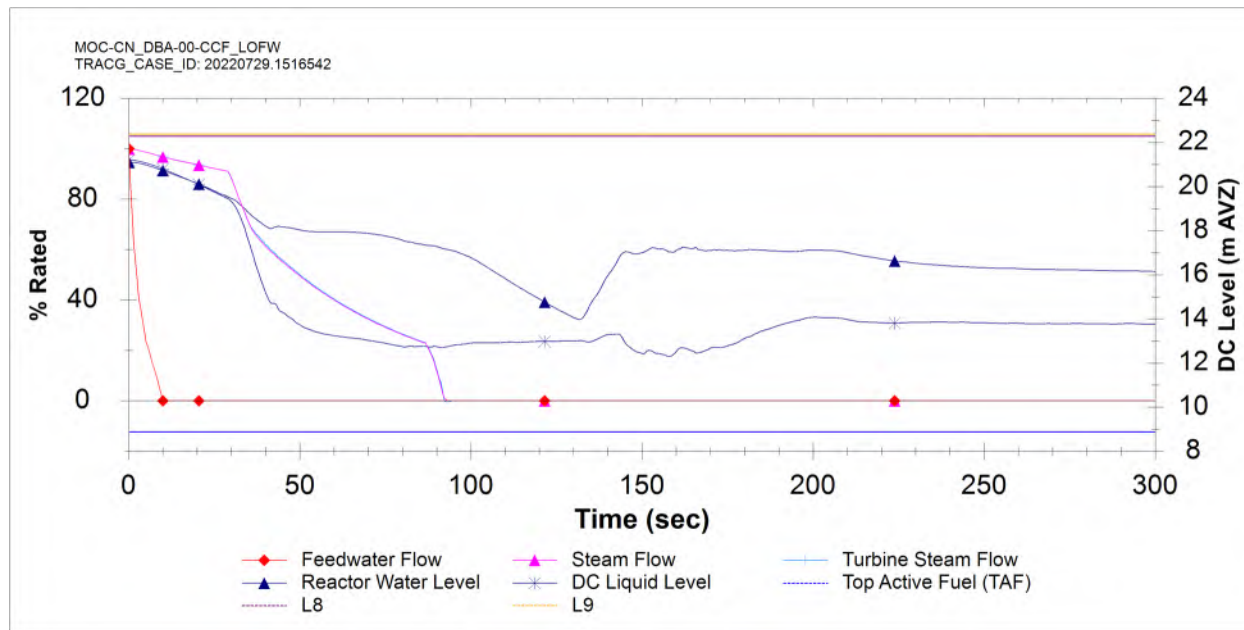


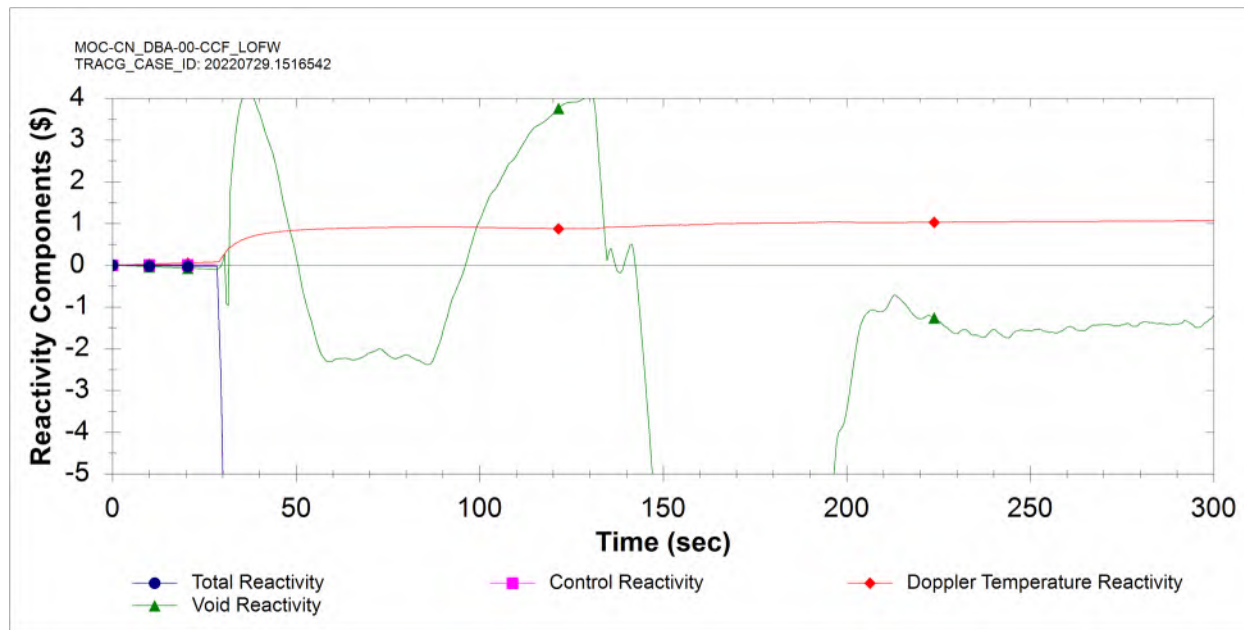
Figure 15.5-89: Loss of Feedwater Flow (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-90: Loss of Feedwater Flow (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-91: Loss of Feedwater Flow (DBA)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

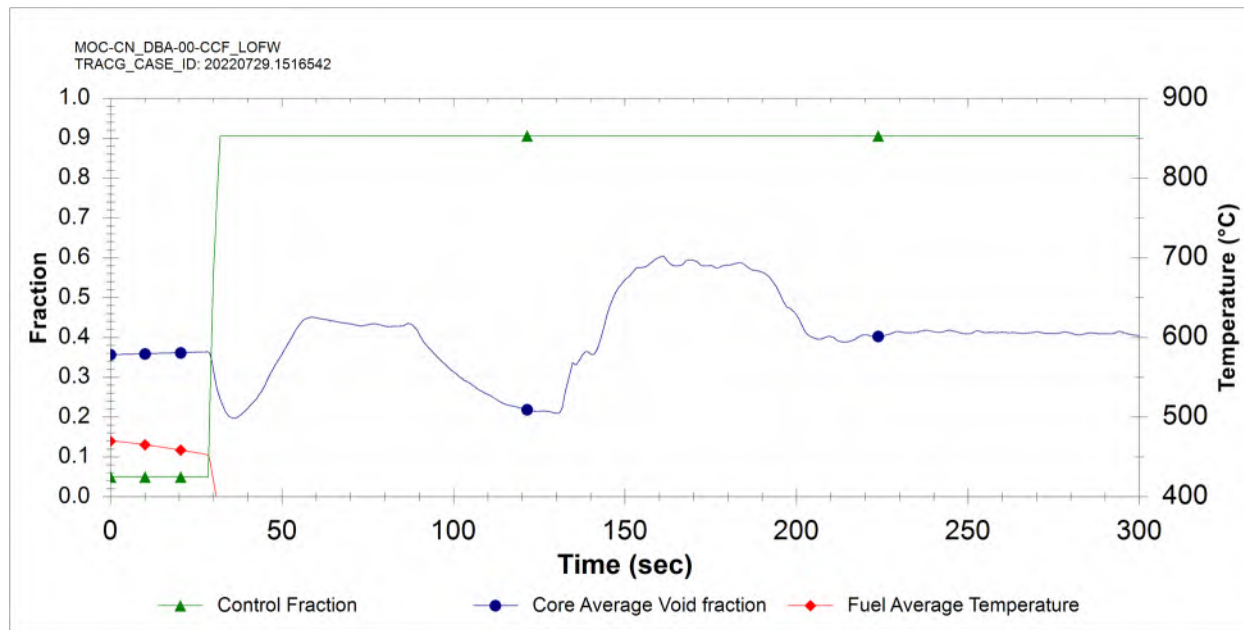


Figure 15.5-92: Loss of Feedwater Flow (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

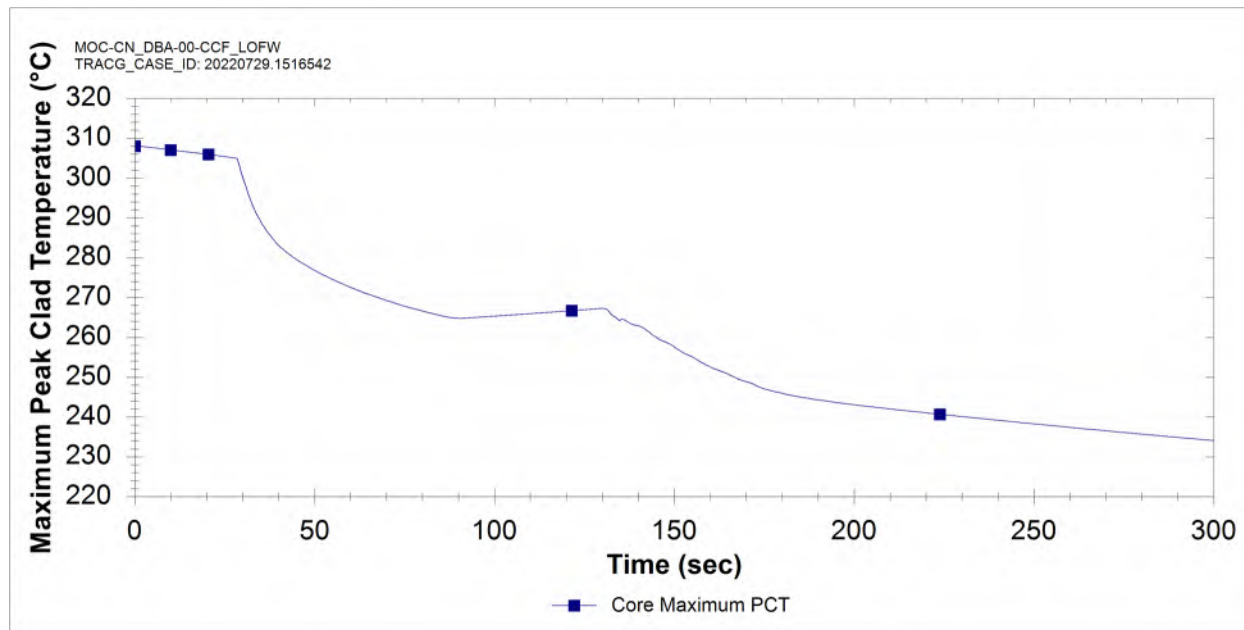


Figure 15.5-93: Loss of Feedwater Flow (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

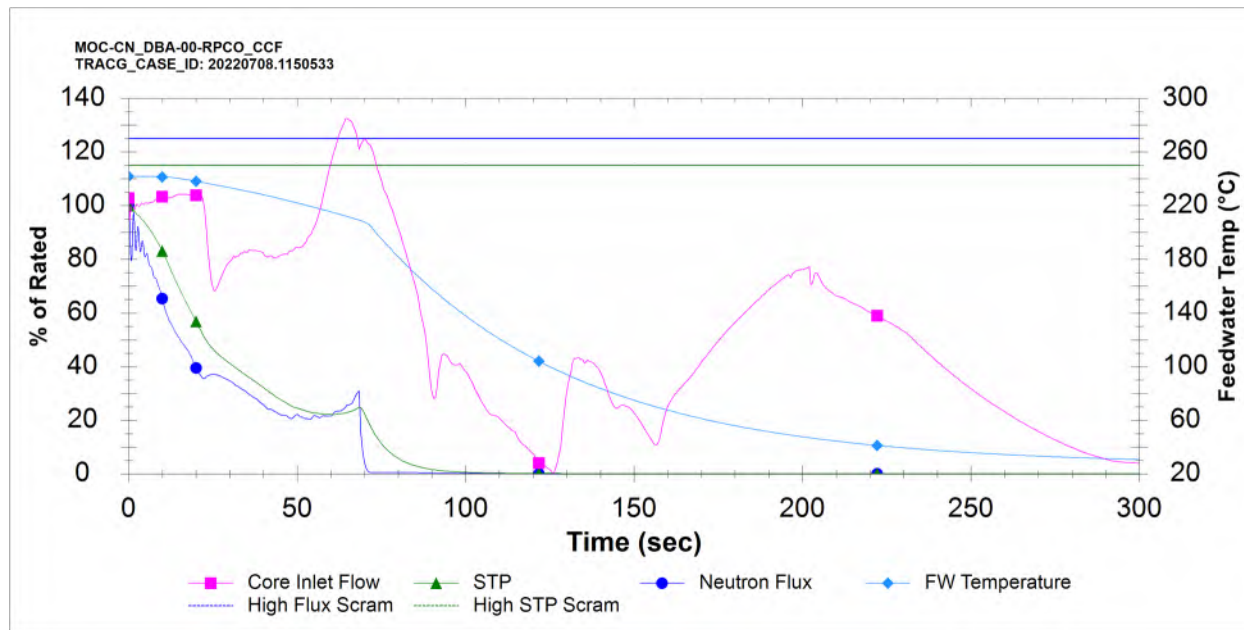


Figure 15.5-94: RPV Pressure Control Open (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

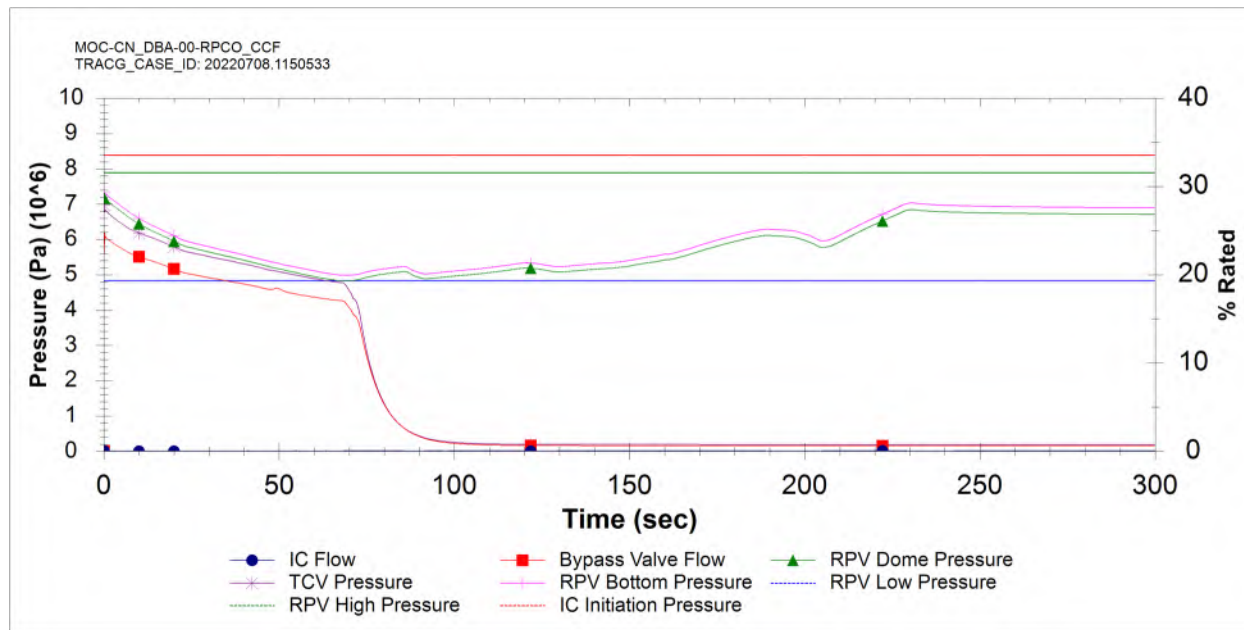


Figure 15.5-95: RLV Pressure Control Open (DBA)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

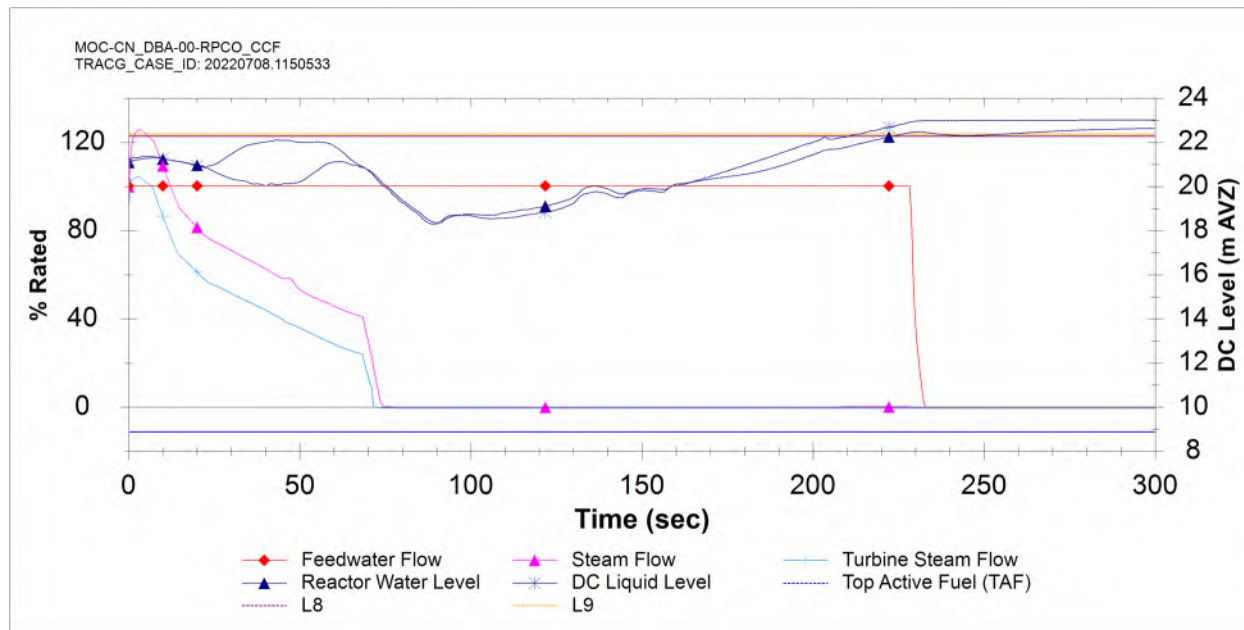


Figure 15.5-96: RPV Pressure Control Open (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

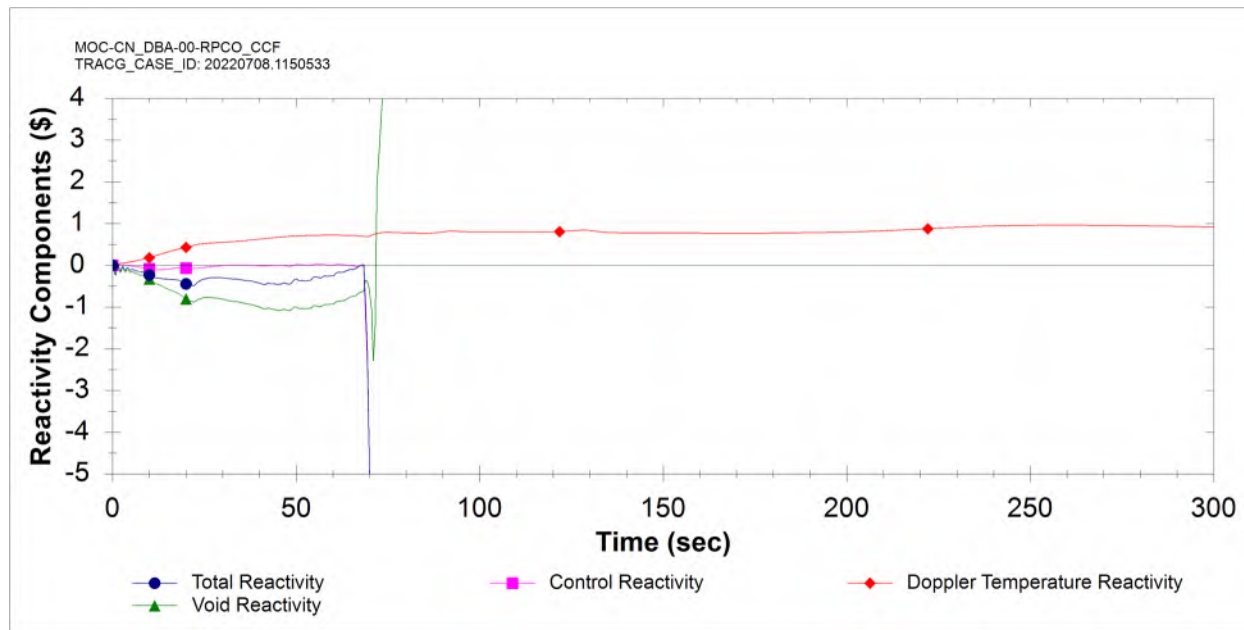


Figure 15.5-97: RPV Pressure Control Open (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

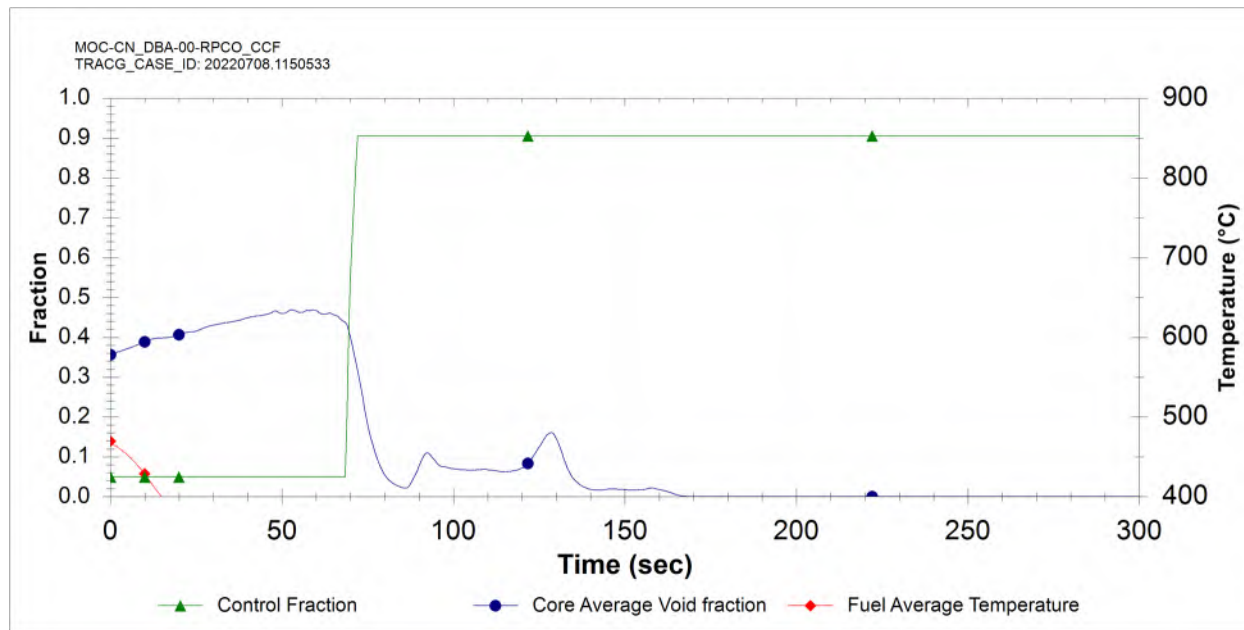


Figure 15.5-98: RPV Pressure Control Open (DBA)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

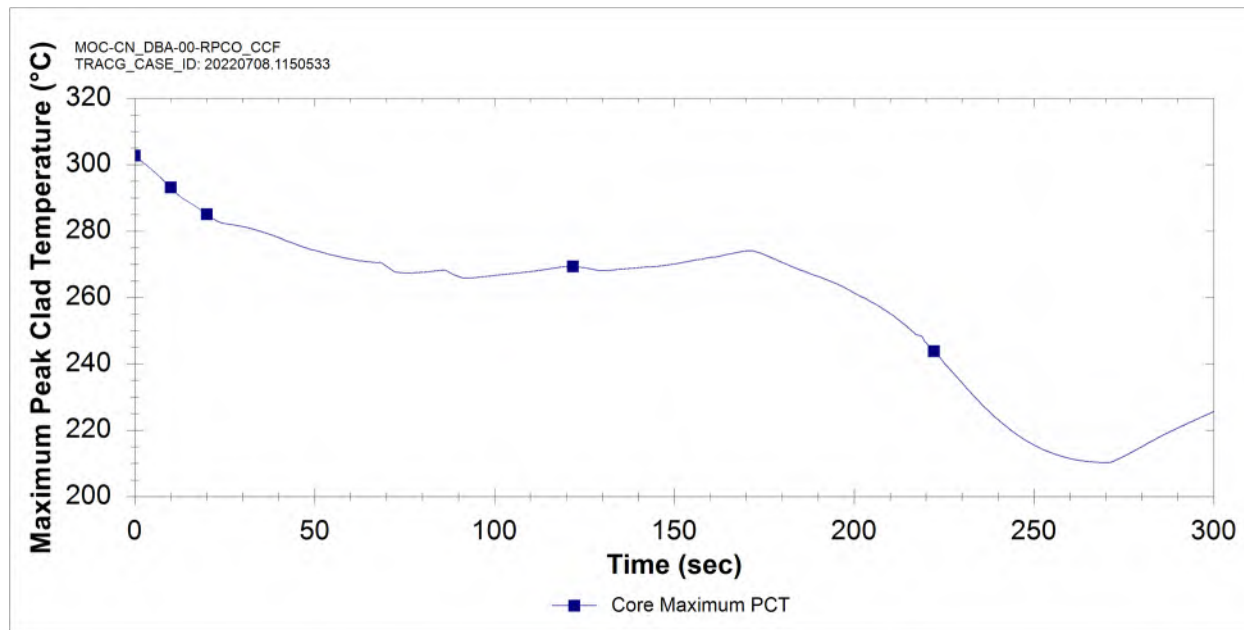


Figure 15.5-99: RPV Pressure Control Open (DBA)

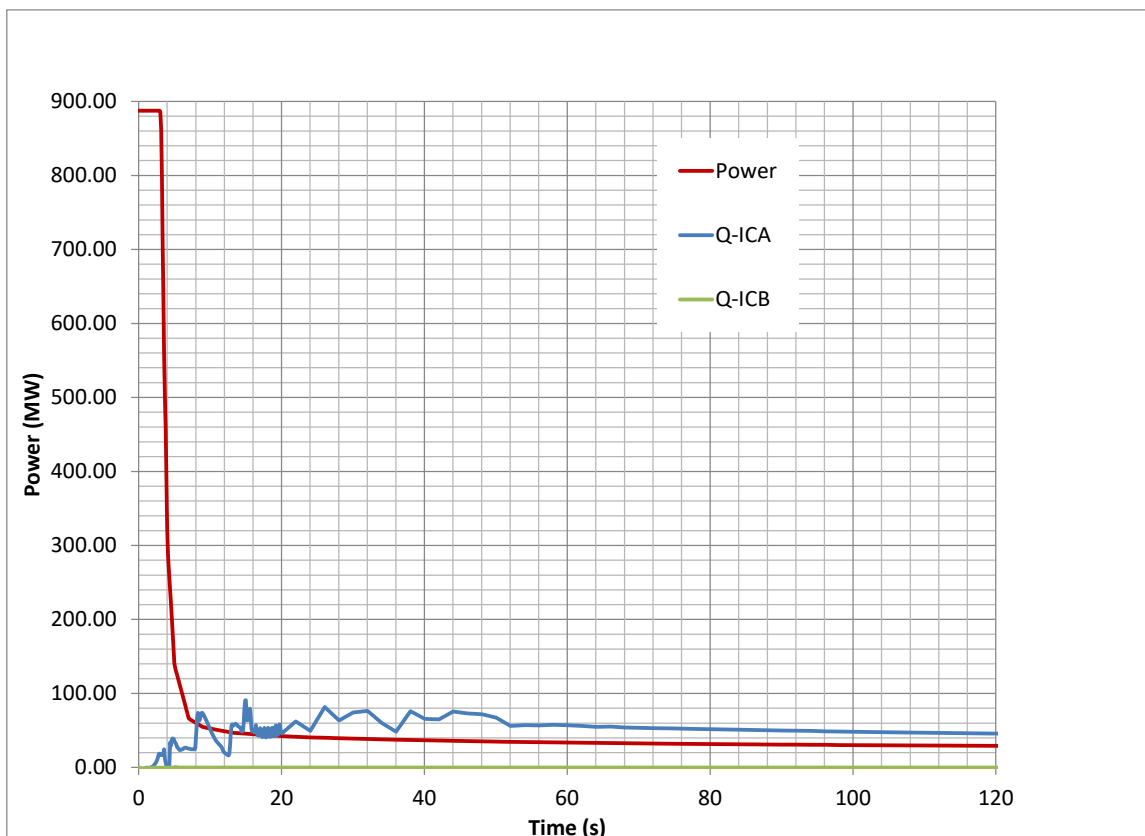
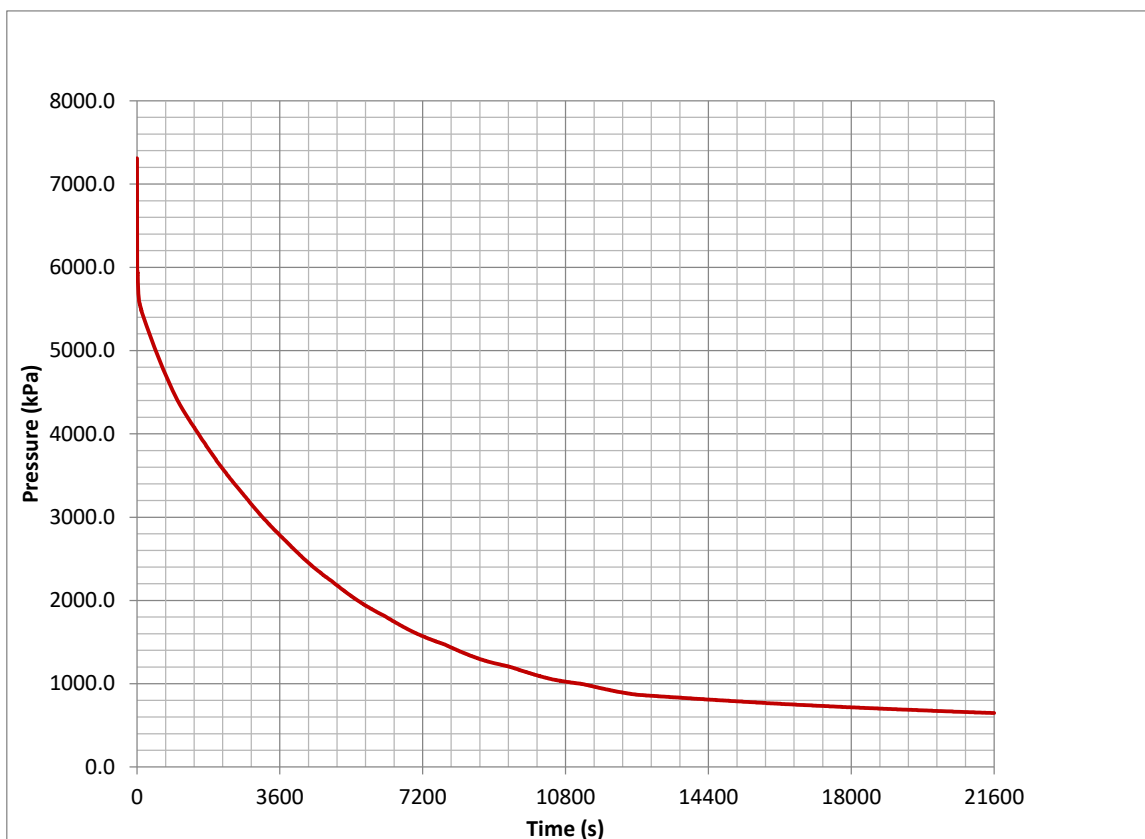
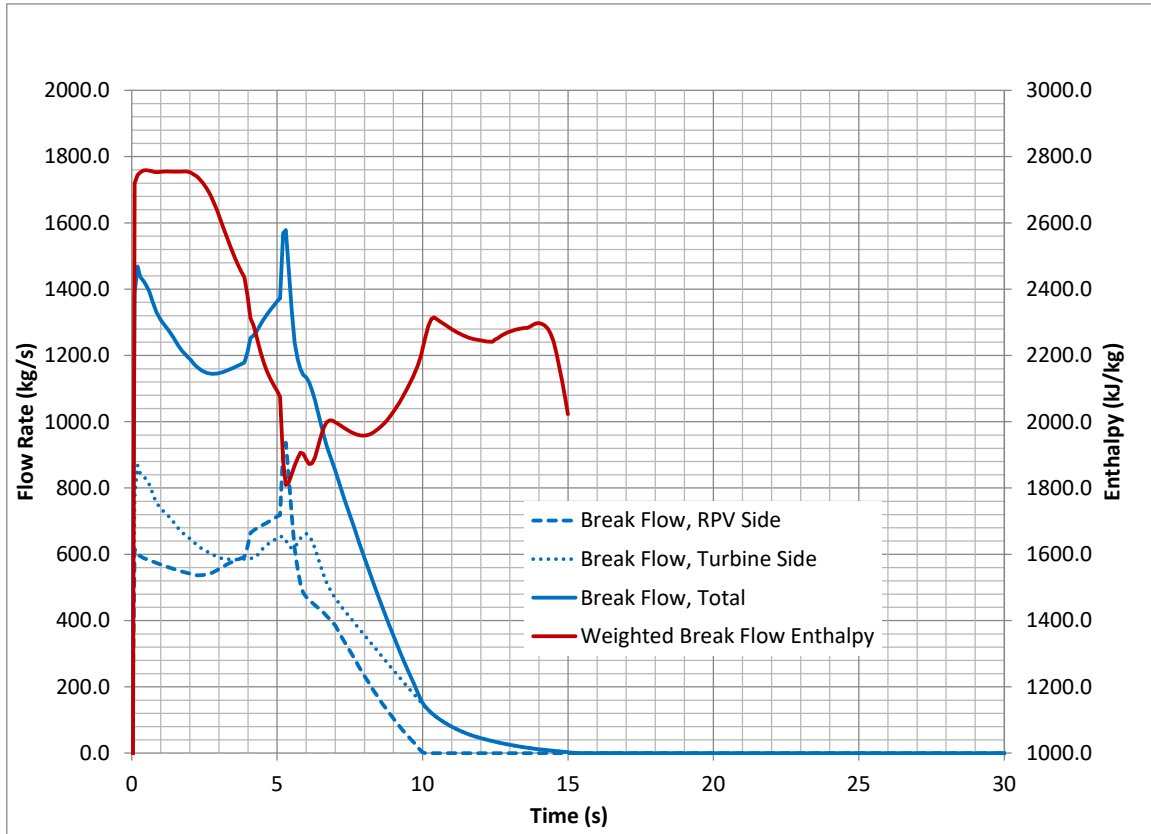


Figure 15.5-100: Reactor Power, Large Main Steam Pipe Break, Conservative Case



**Figure 15.5-101: Reactor Pressure, Large Main Steam Pipe Break, Conservative Case**



**Figure 15.5-102: Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case**

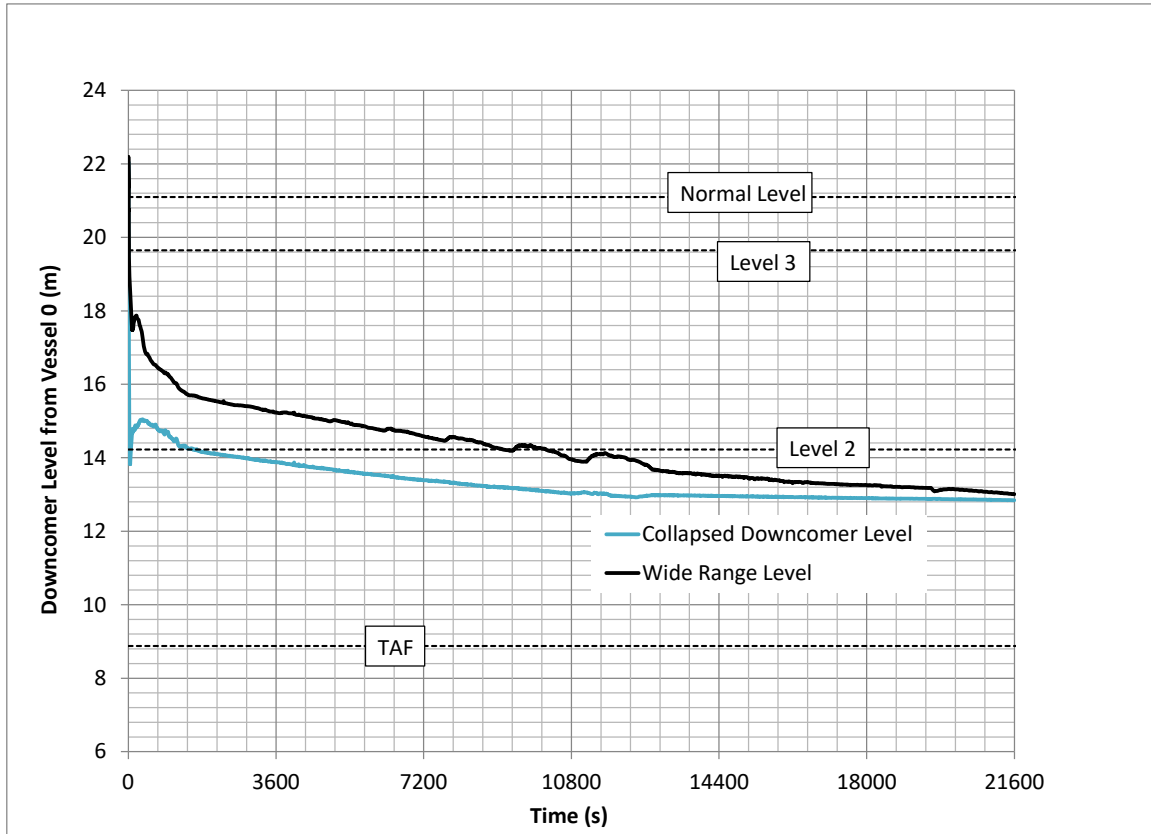
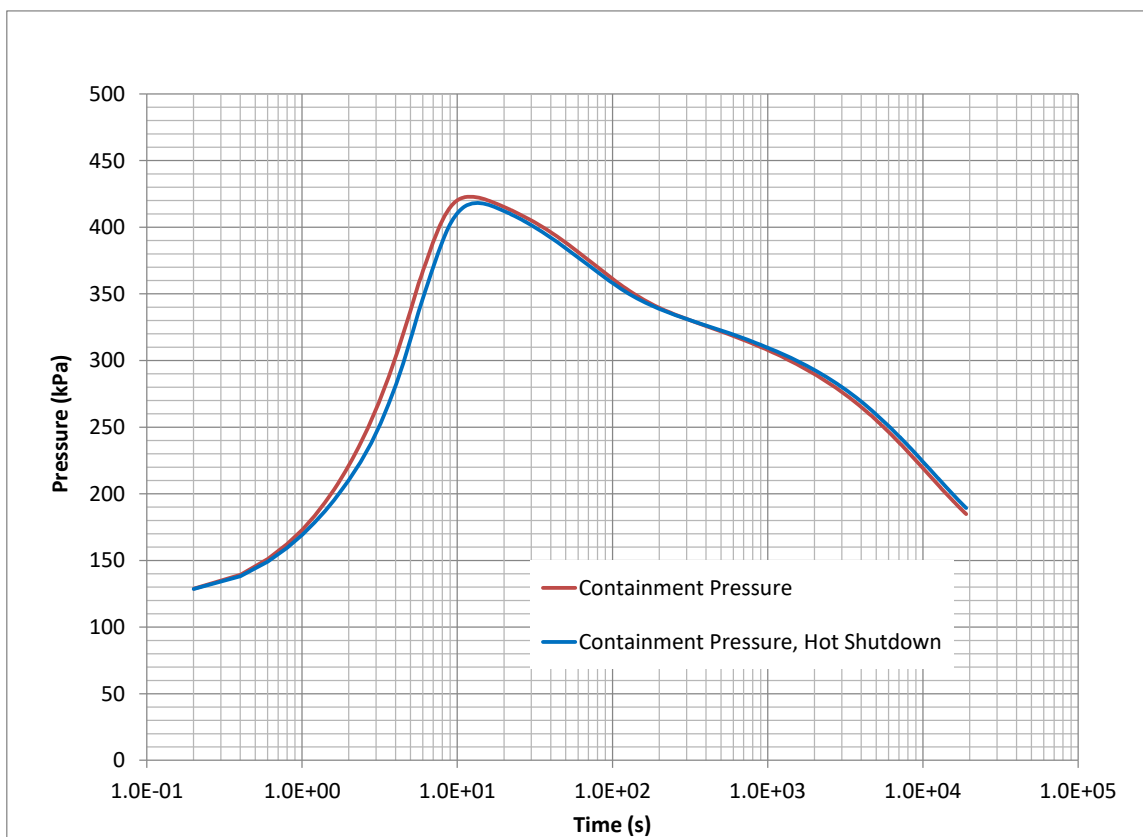
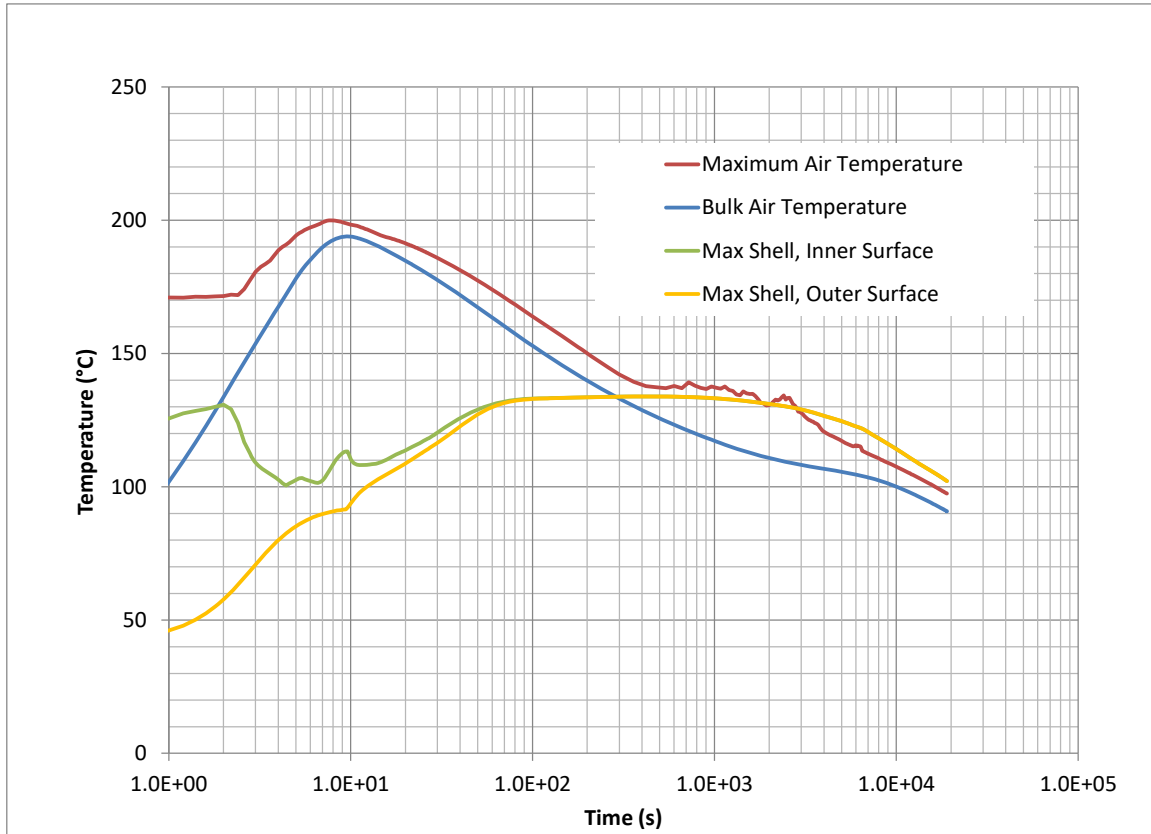


Figure 15.5-103: Reactor Water Level, Large Main Steam Pipe Break, Conservative Case





**Figure 15.5-104: Containment Pressure, Large Main Steam Pipe Break, Conservative Case**



**Figure 15.5-105: Containment Temperatures, Large Main Steam Pipe Break, Conservative Case**

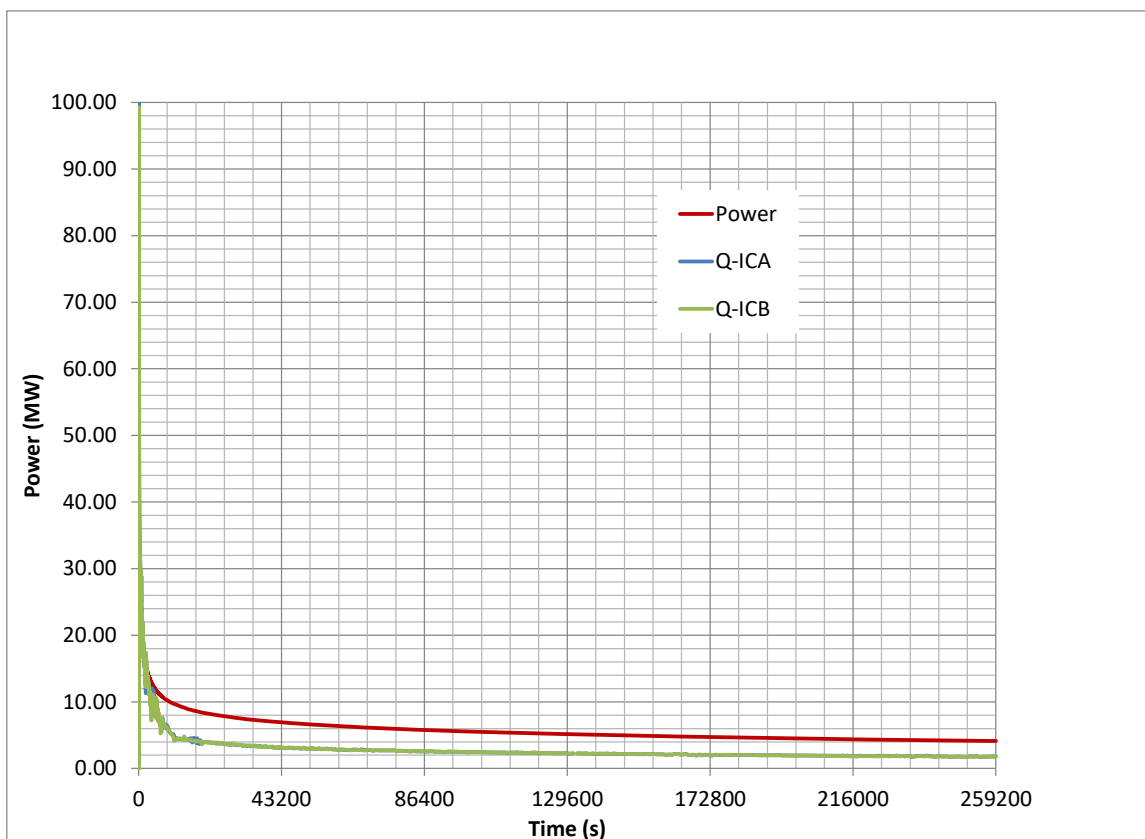
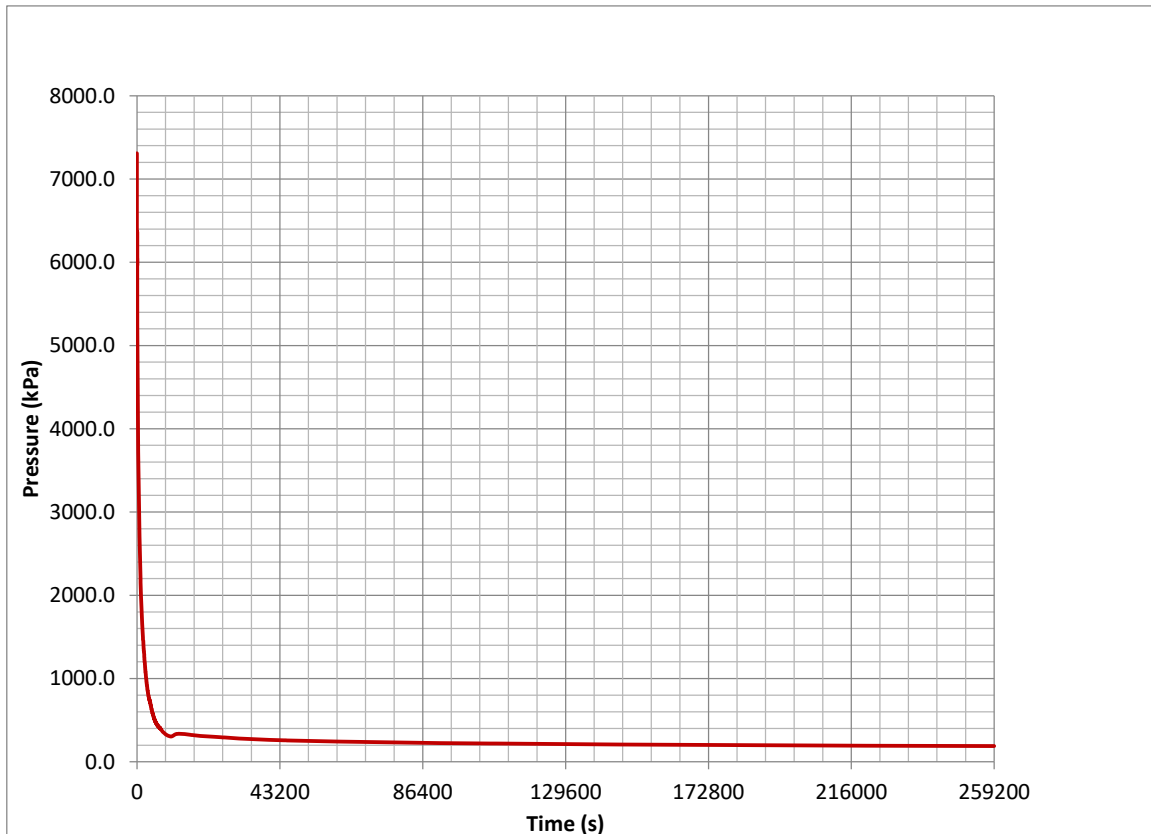
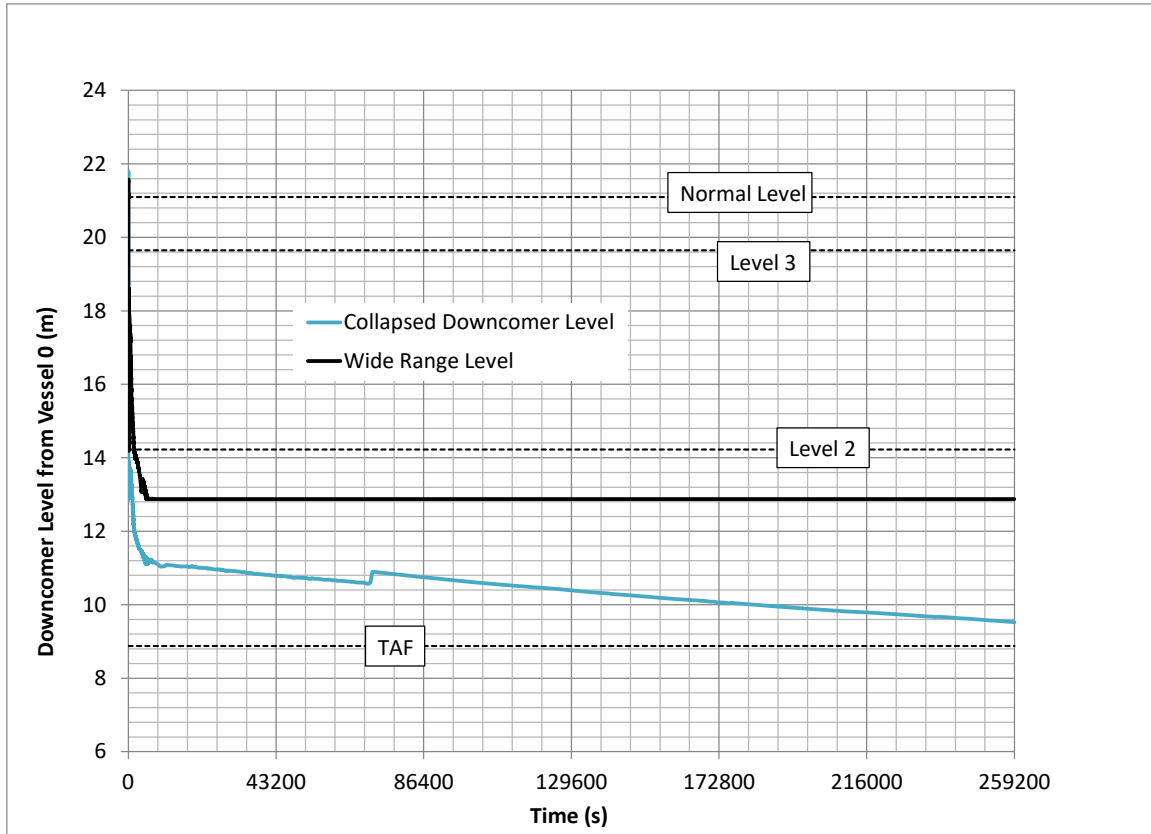


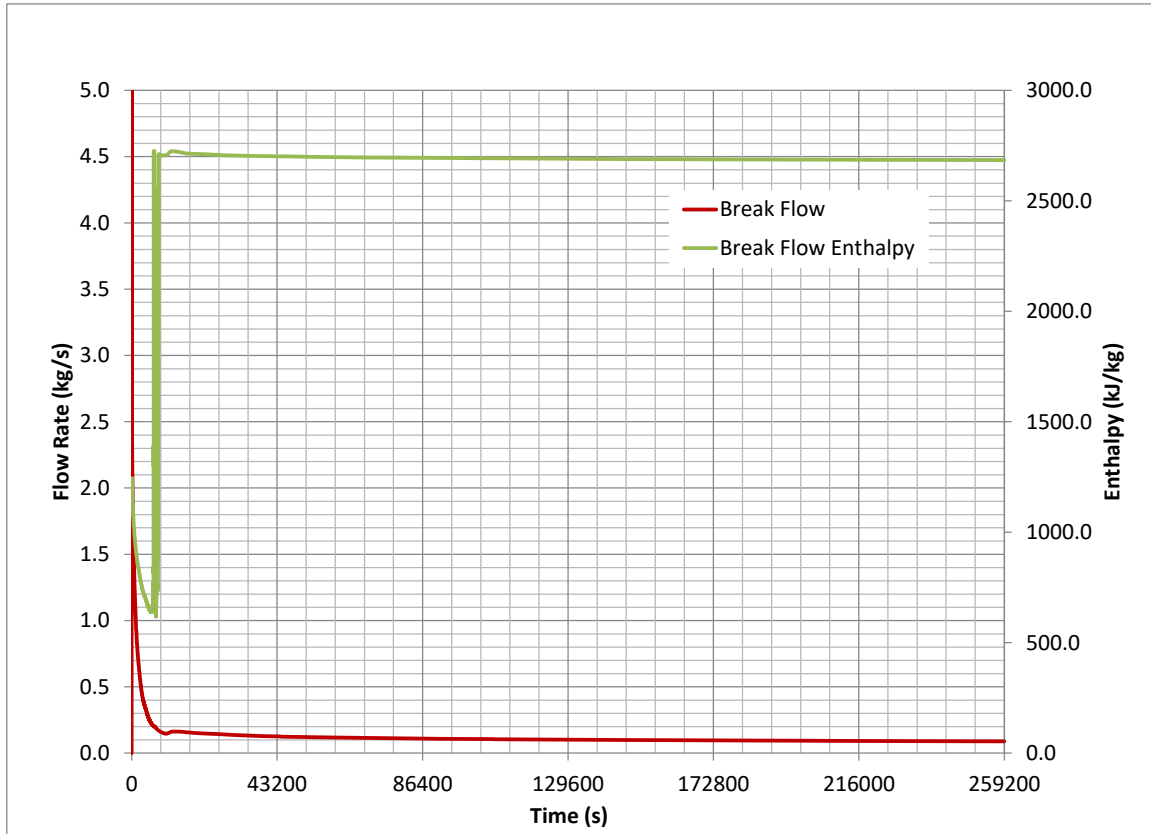
Figure 15.5-106: Reactor Power, Small Steam Break With LOPP, Conservative Case



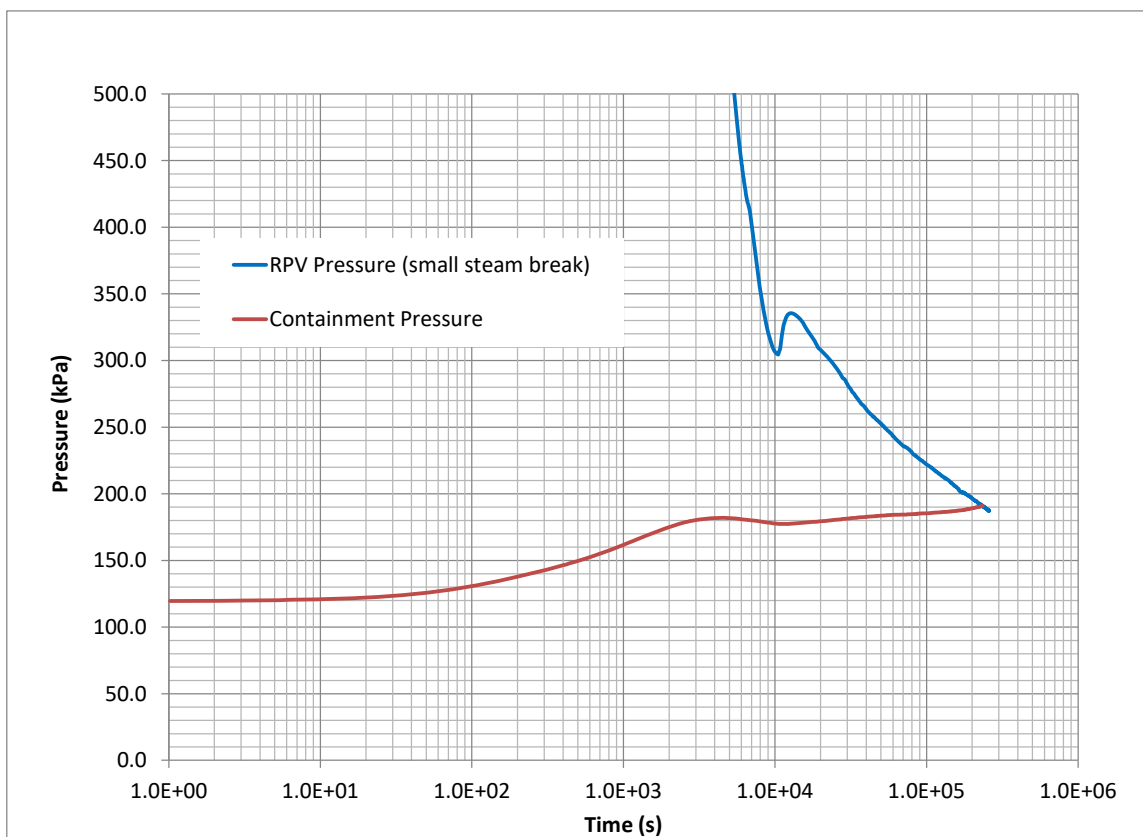
**Figure 15.5-107: Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case**



**Figure 15.5-108: Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case**



**Figure 15.5-109: Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case**



**Figure 15.5-110: Containment Pressure, Small Steam Pipe Break with LOPP 2 ICS Trains, Conservative Case**

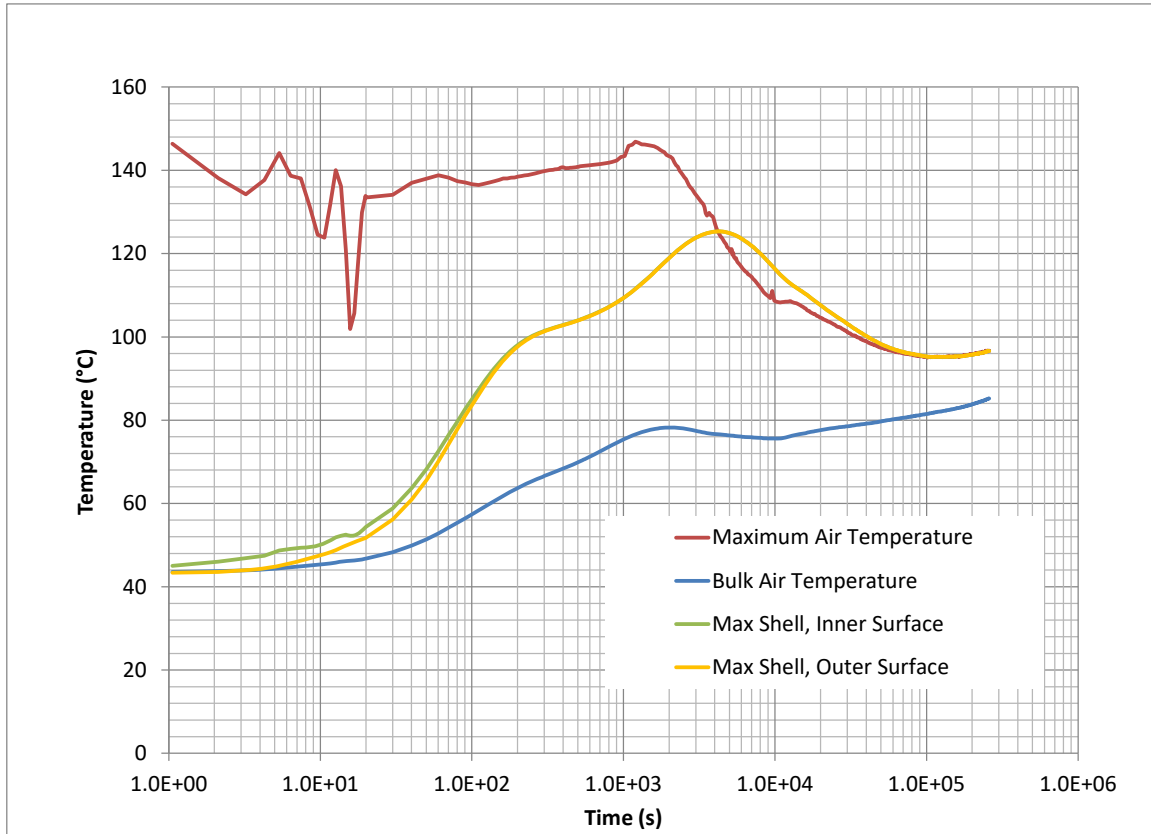


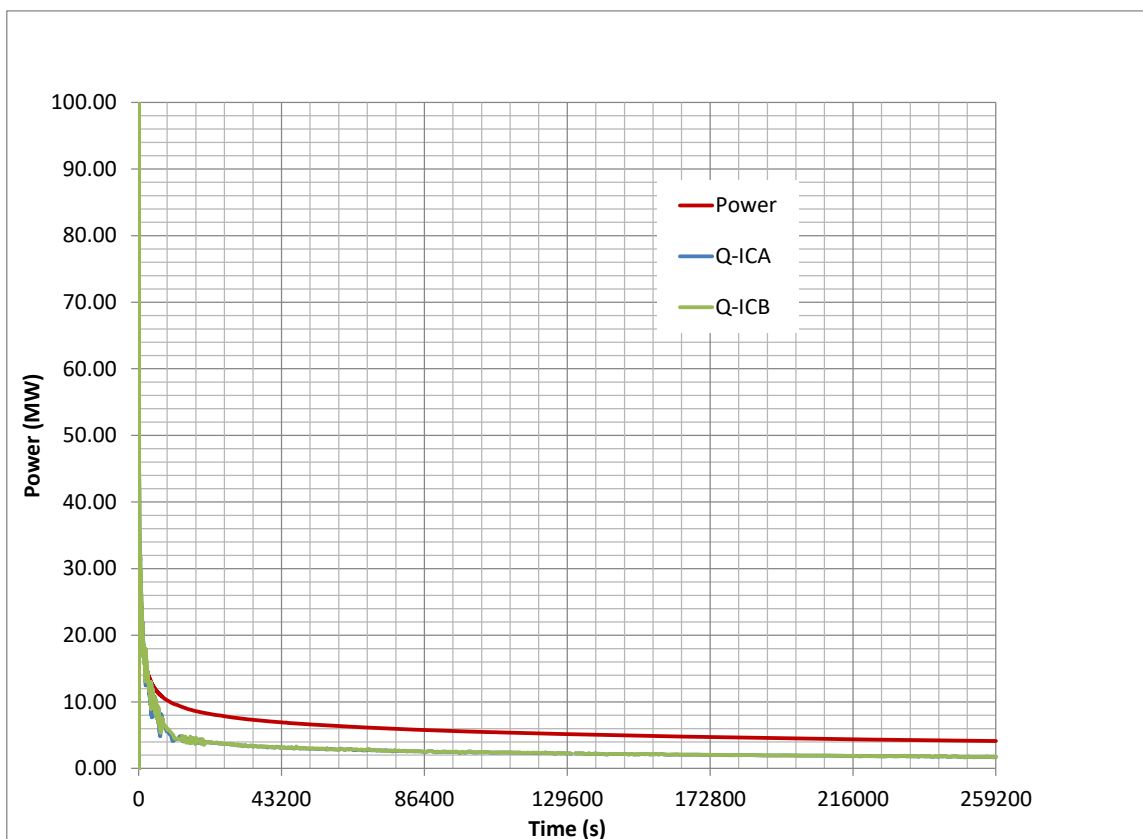
Figure 15.5-111: Containment Temperature, Small Steam Pipe Break, Conservative Case



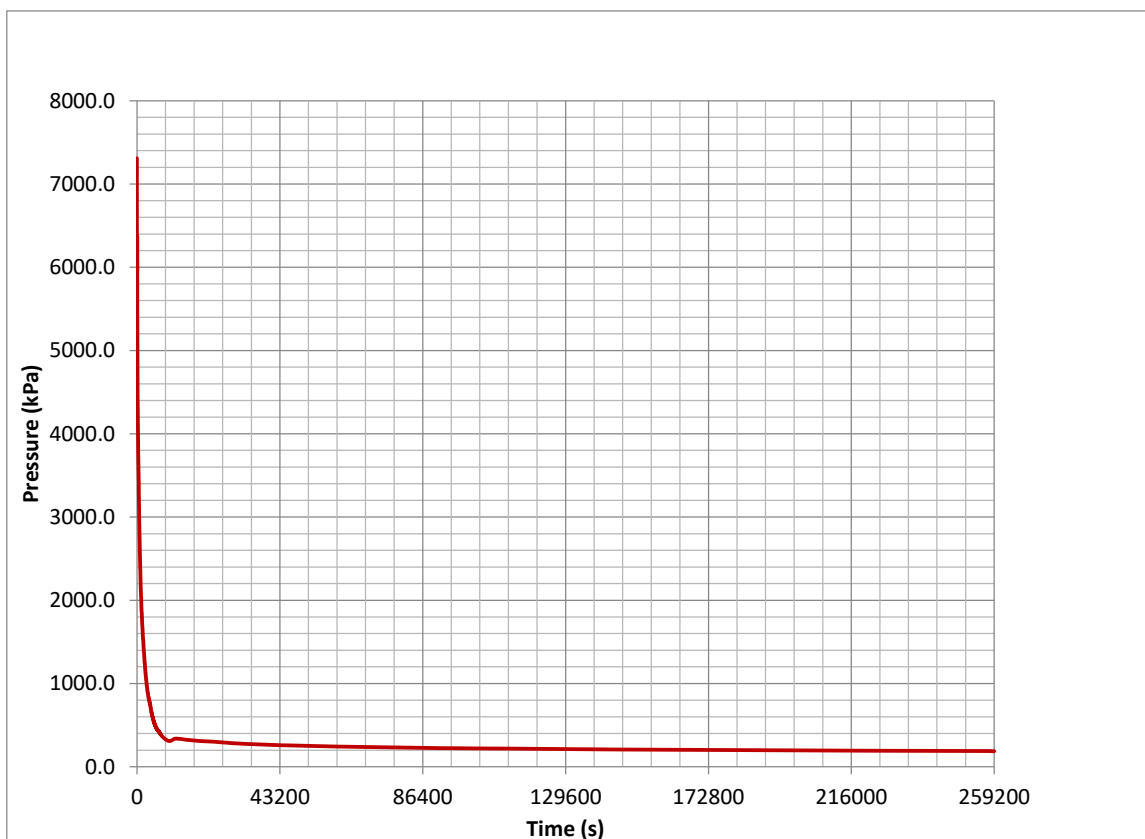
**Figure 15.5-112: Not Used**

**Figure 15.5-113: Not Used**

**Figure 15.5-114: Not Used**



**Figure 15.5-115: Reactor Power, Small Liquid Pipe Break, Conservative Case**



**Figure 15.5-116: Reactor Pressure, Small Liquid Pipe Break, Conservative Case**

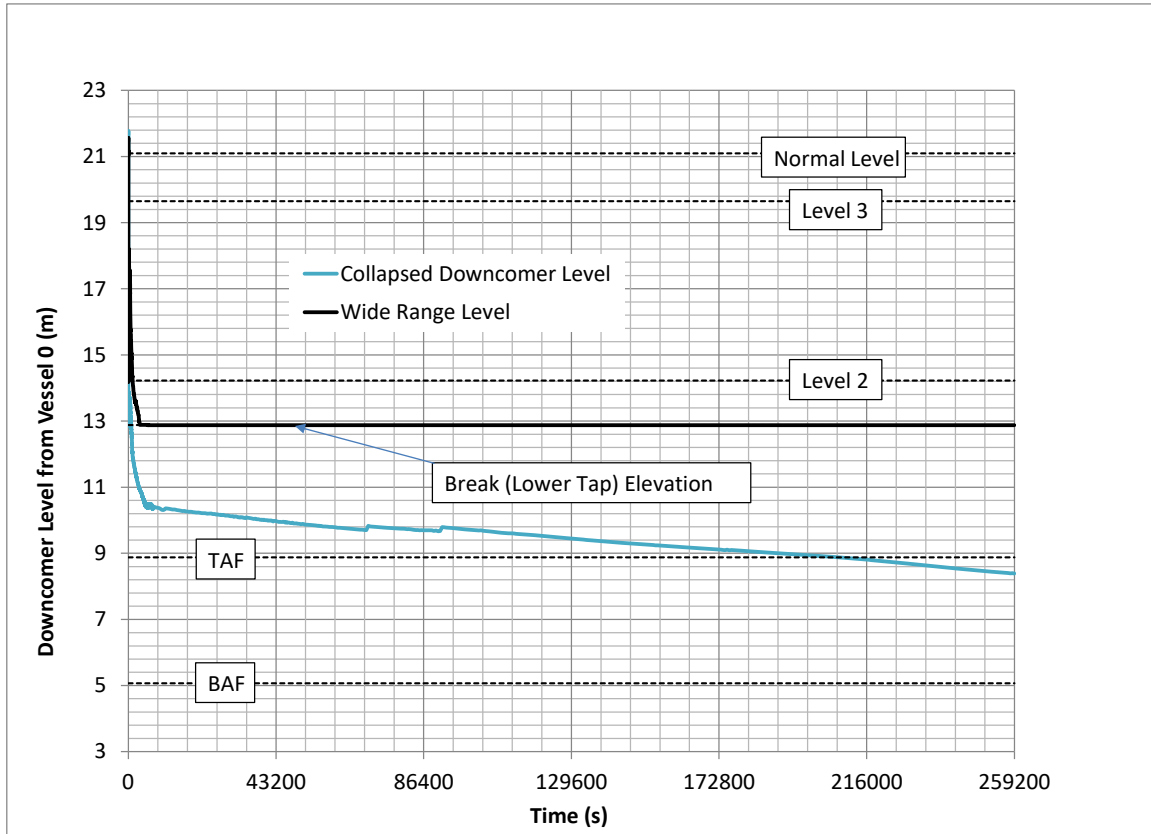
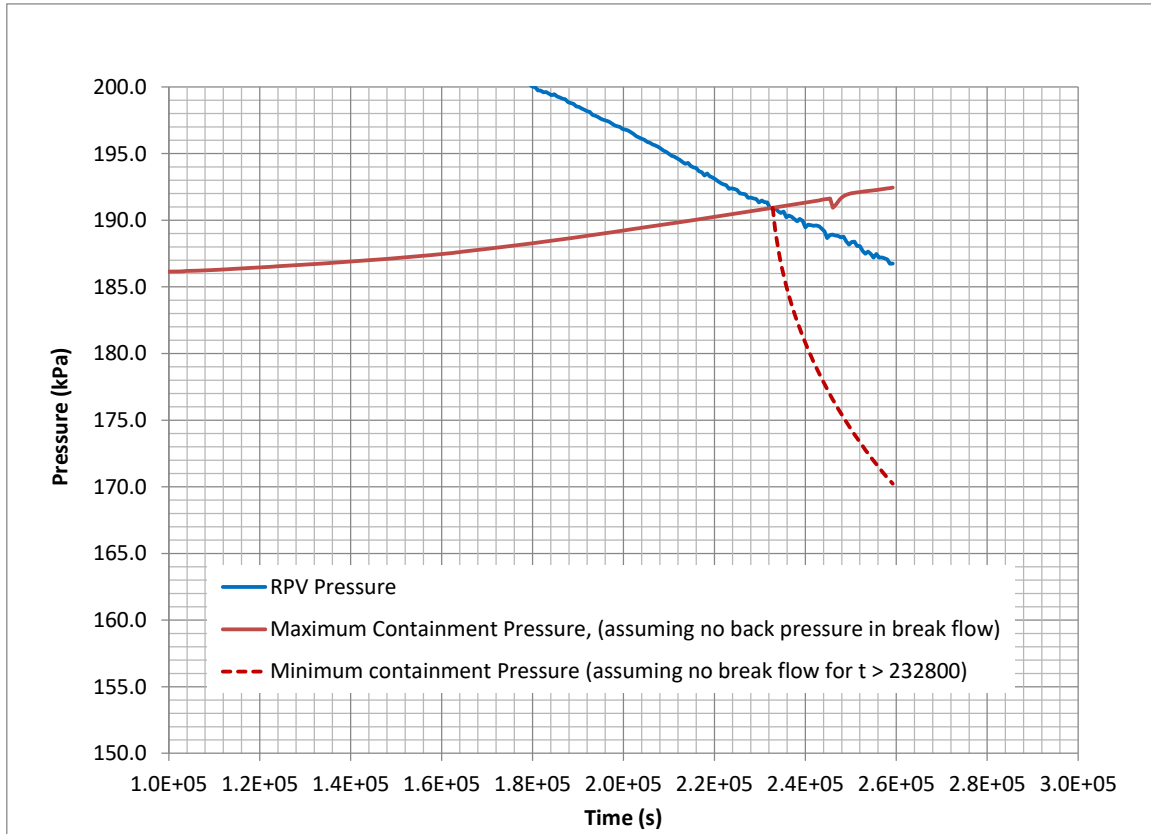
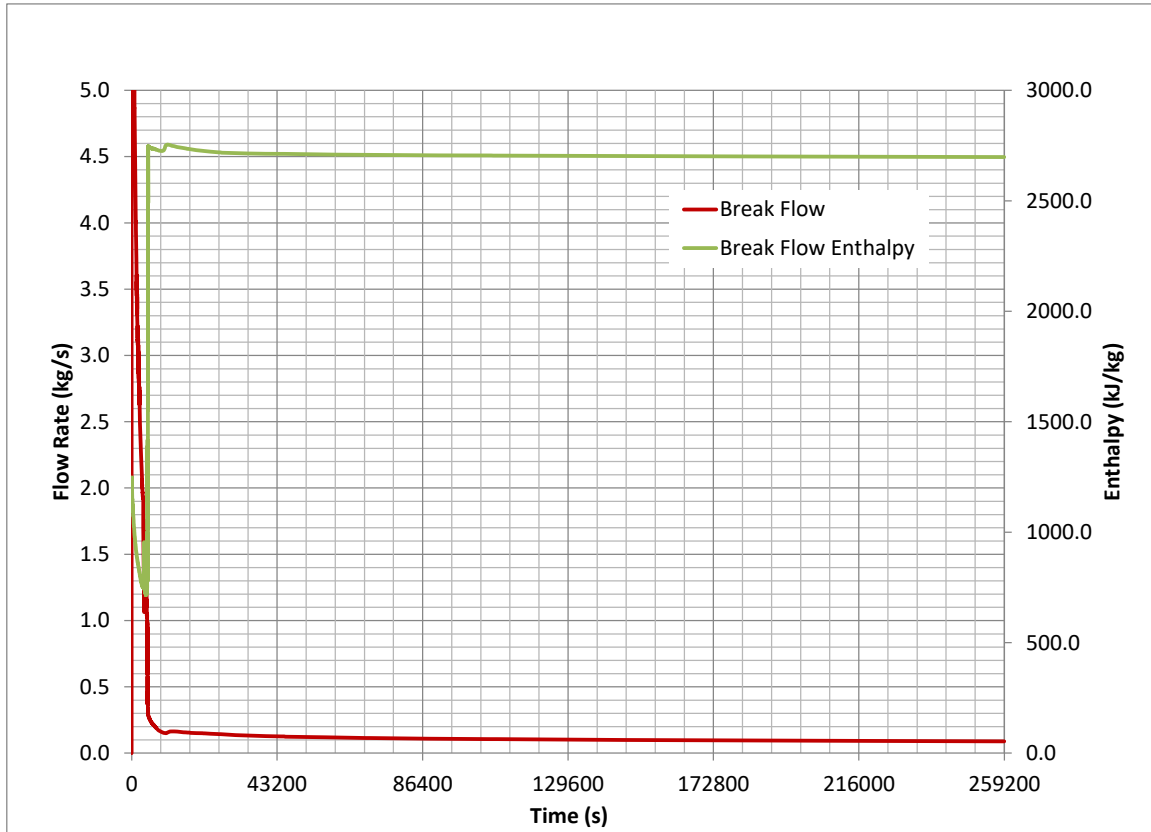


Figure 15.5-117: Reactor Water Level, Small Liquid Break, Conservative Case



**Figure 15.5-118: Containment Pressure After RPV Depressurization, Small Liquid Pipe Break, Conservative Case**



**Figure 15.5-119: Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case**

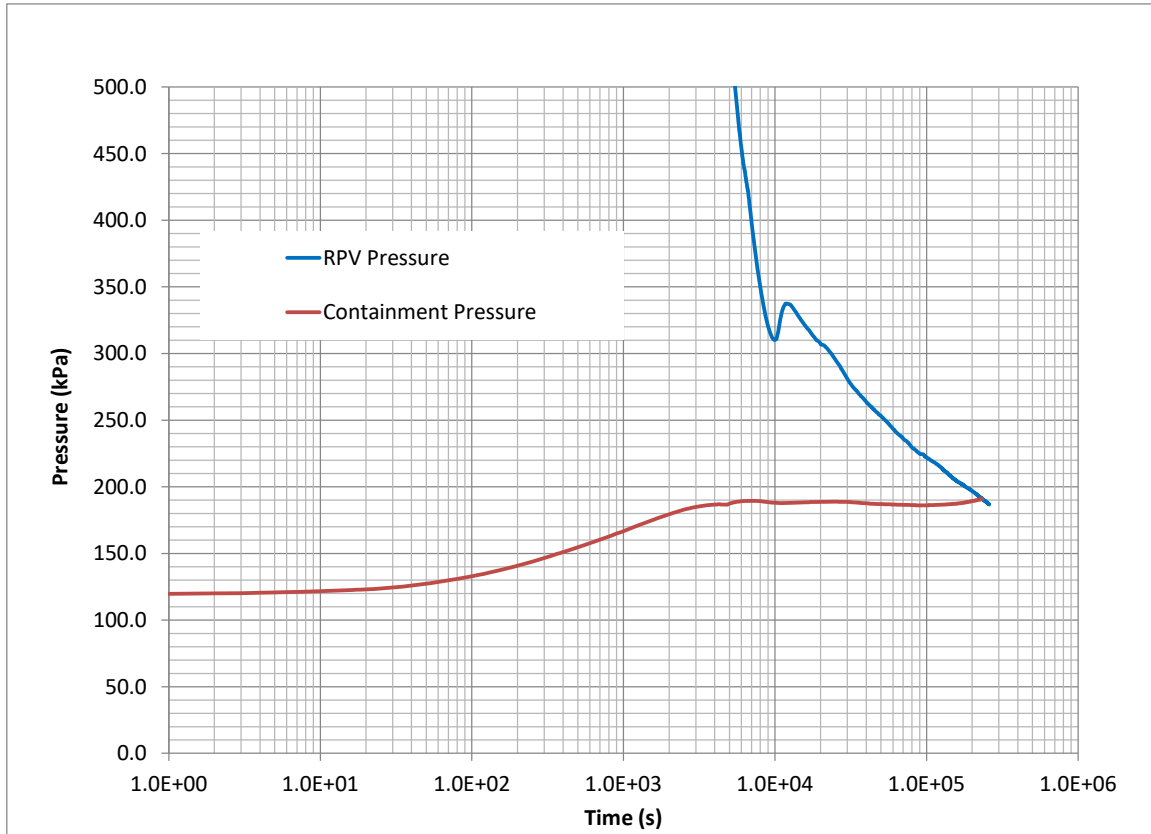


Figure 15.5-120: Containment Pressure, Small Liquid Pipe Break, Conservative Case

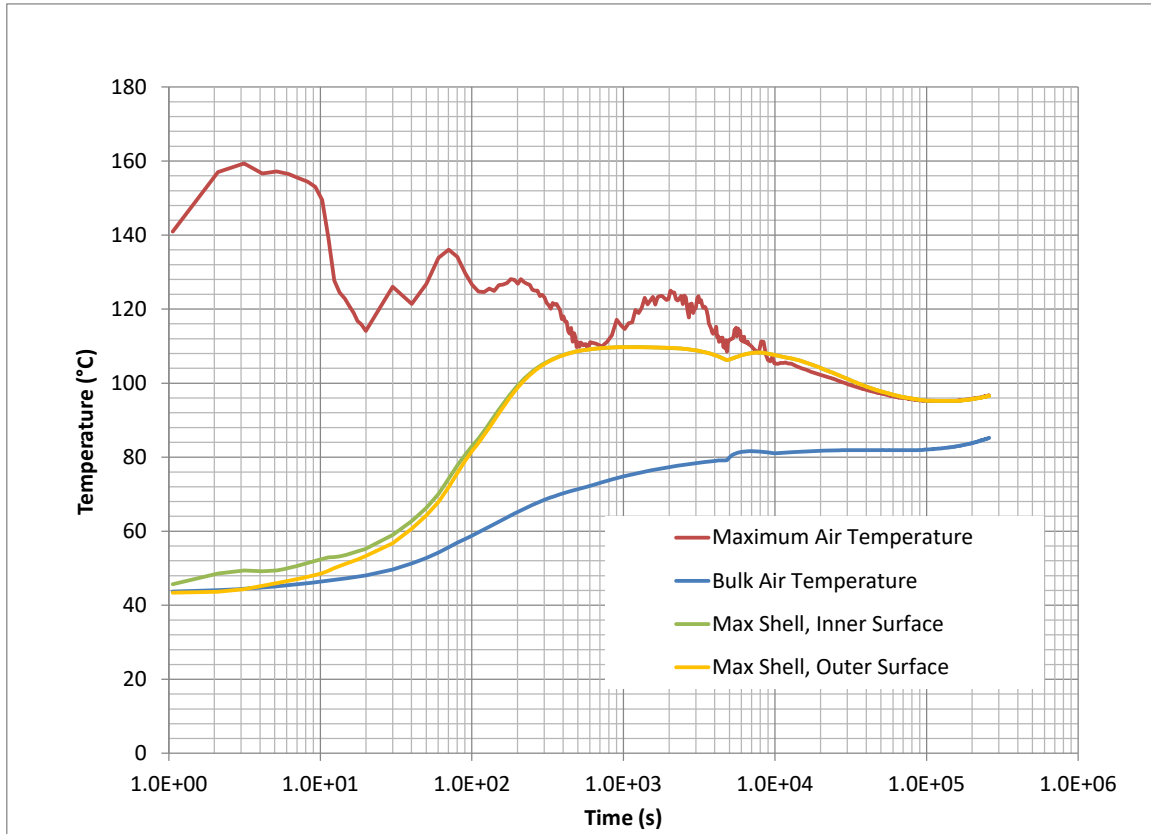
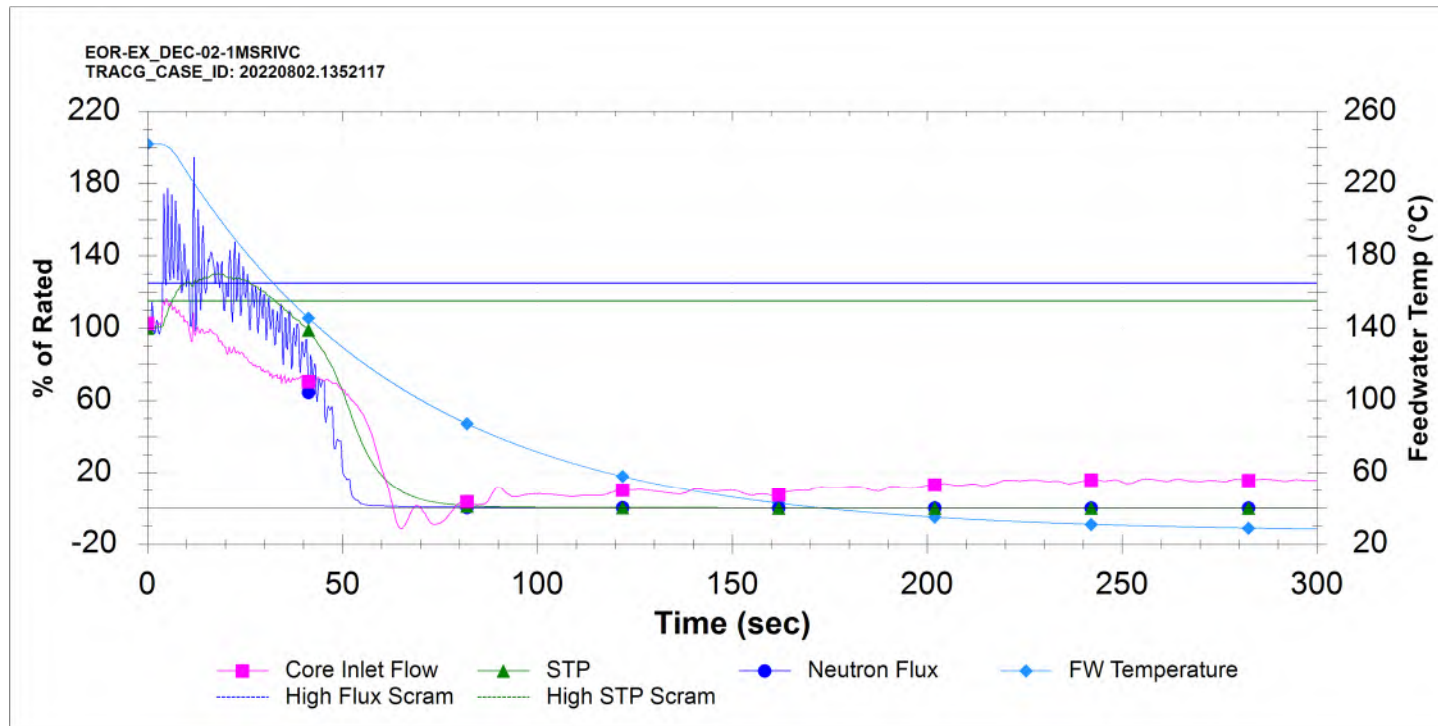


Figure 15.5-121: Containment Temperature, Small Liquid Pipe Break, Conservative Case



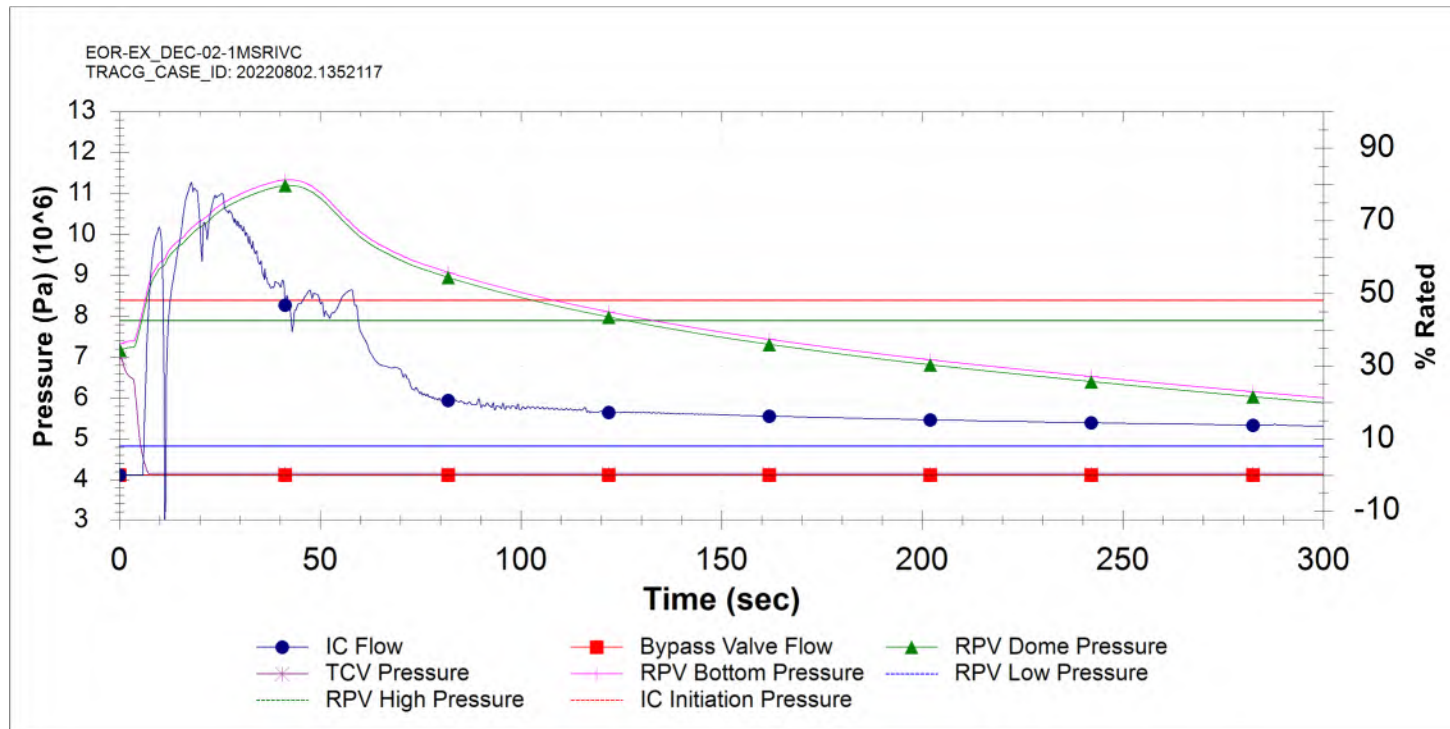
**Figure 15.5-122: Not Used**

**Figure 15.5-123: Not Used**



**Figure 15.5-124: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-125: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

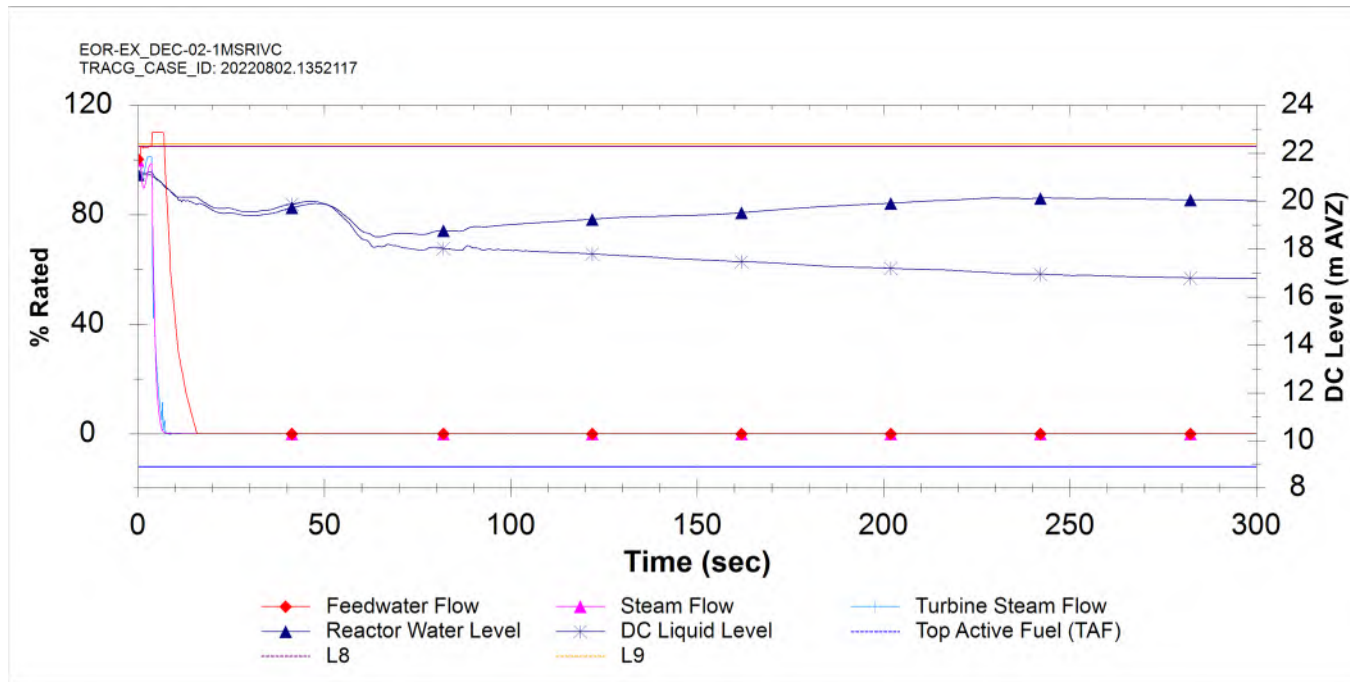
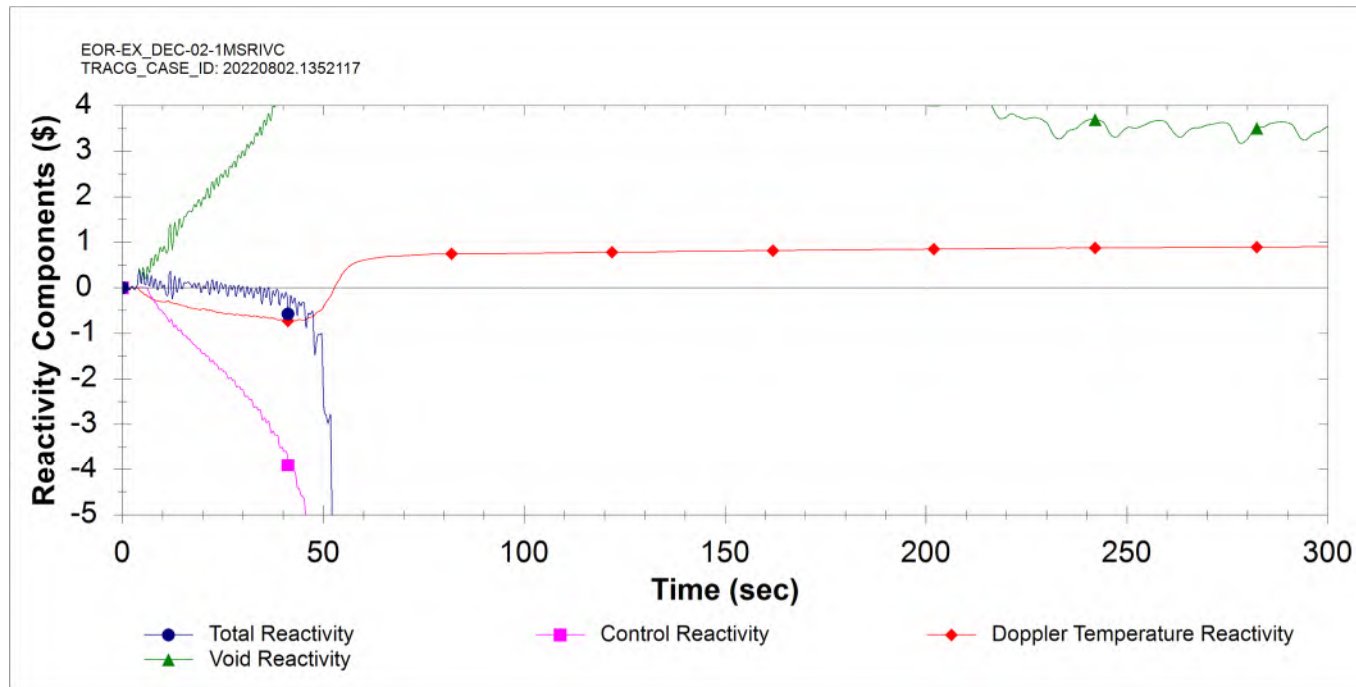


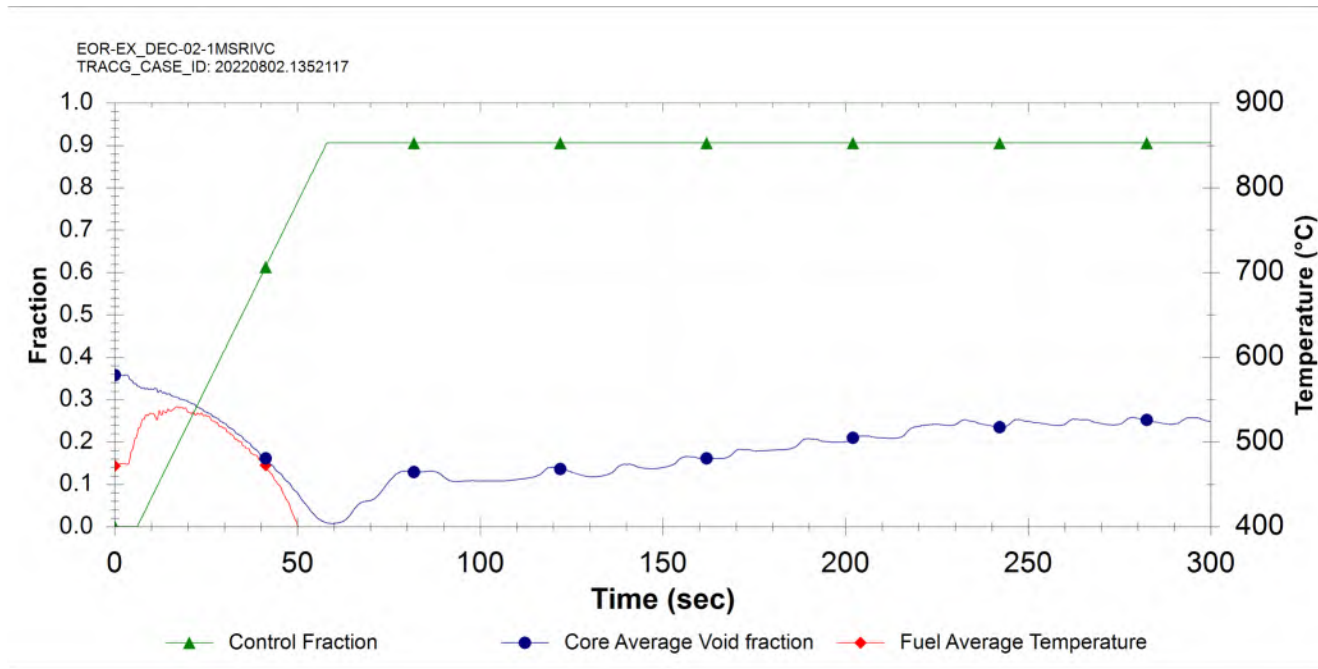
Figure 15.5-126: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



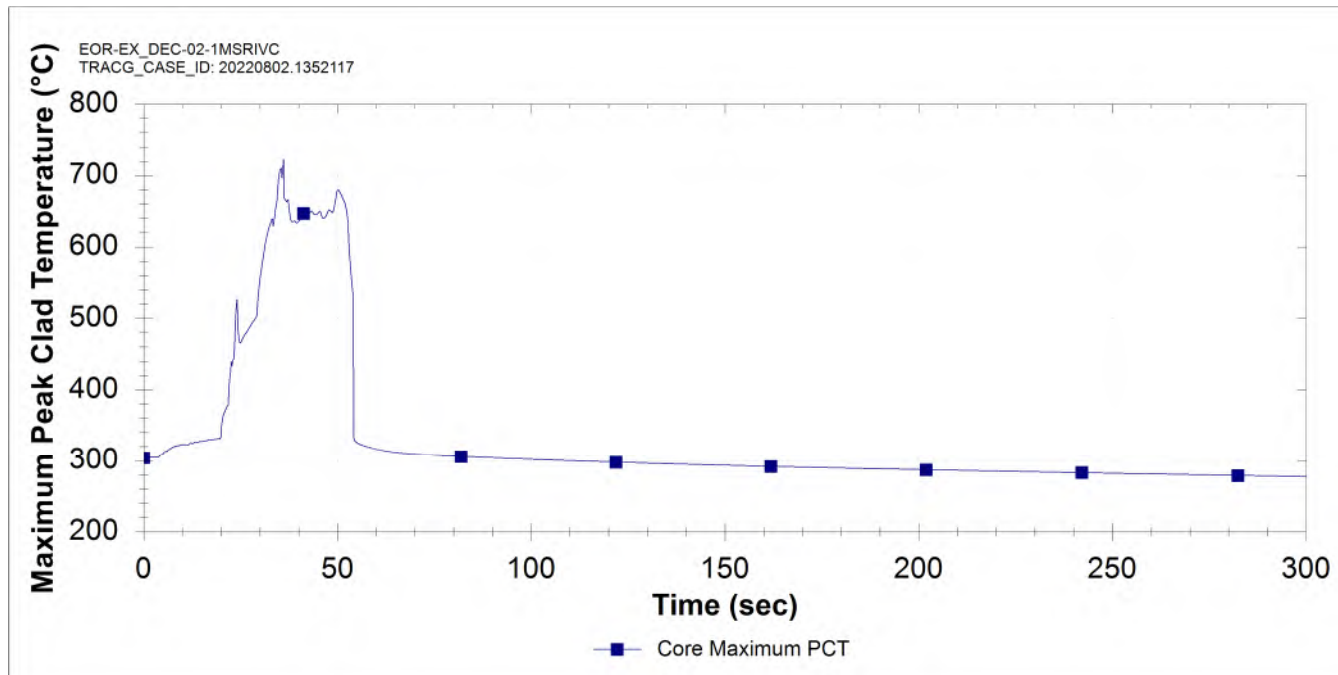
**Figure 15.5-127: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-128: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-129: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

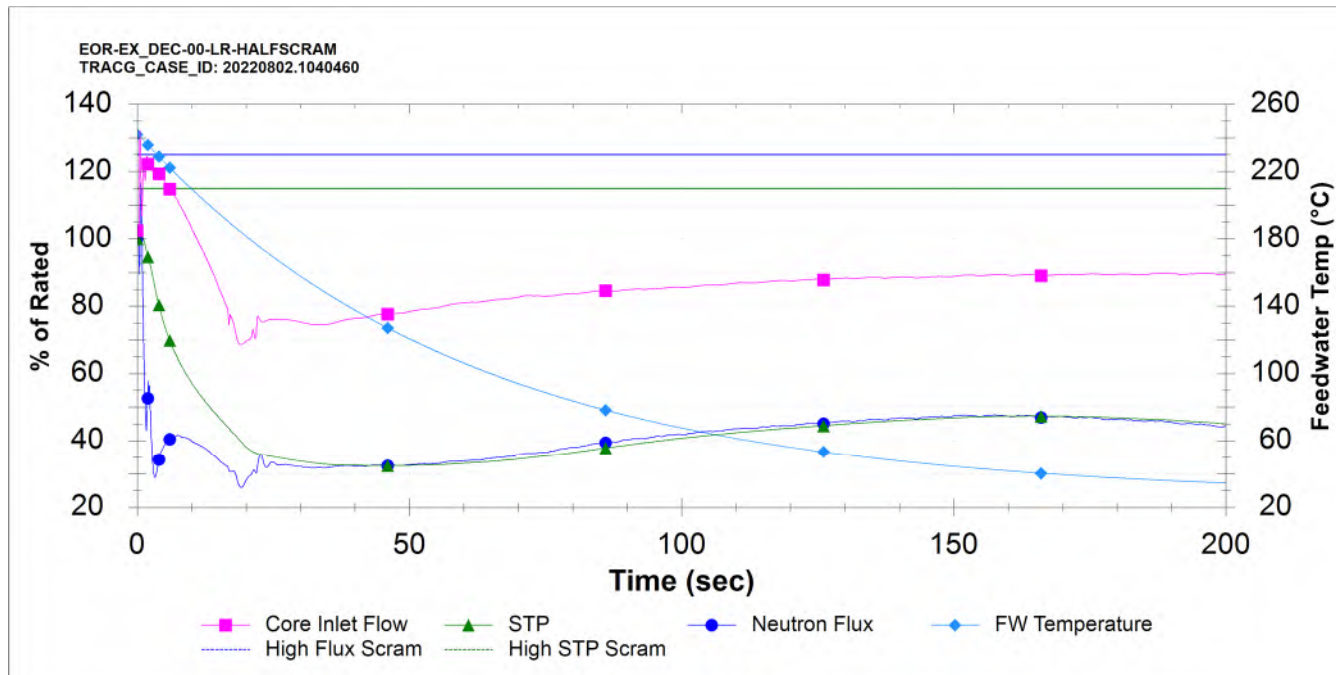


Figure 15.5-130: Complex Sequence Generator Load Rejection (DEC)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

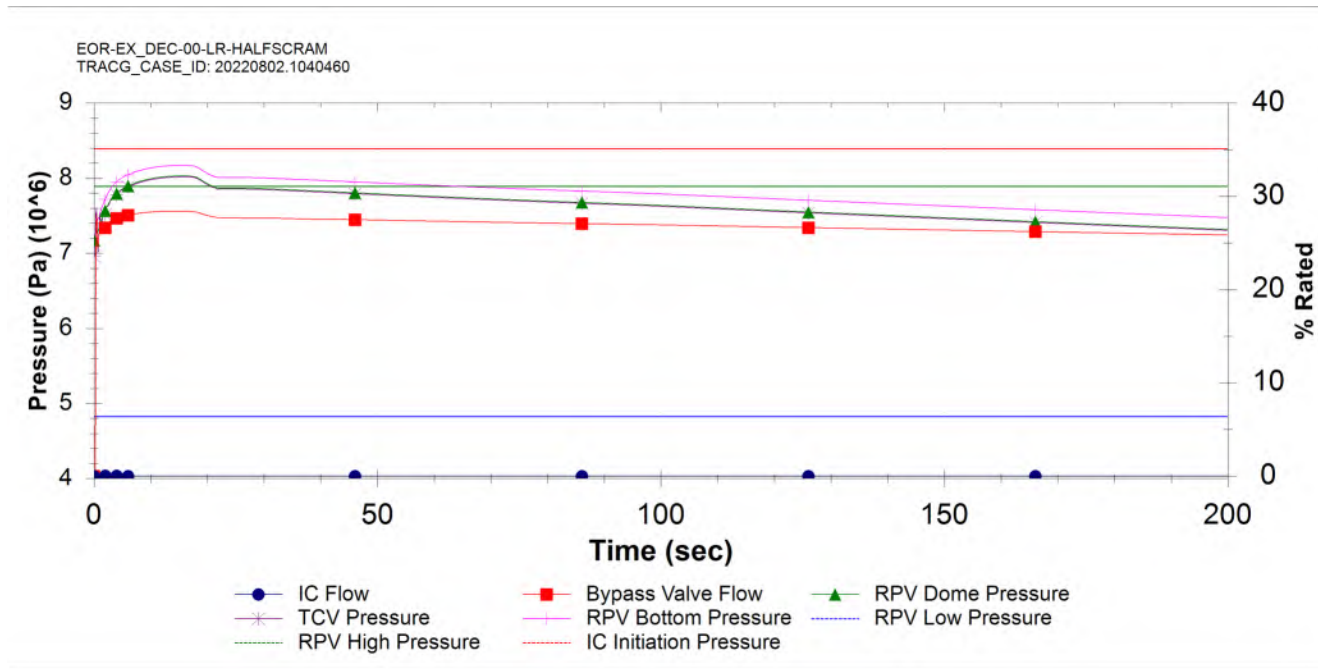


Figure 15.5-131: Complex Sequence Generator Load Rejection (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

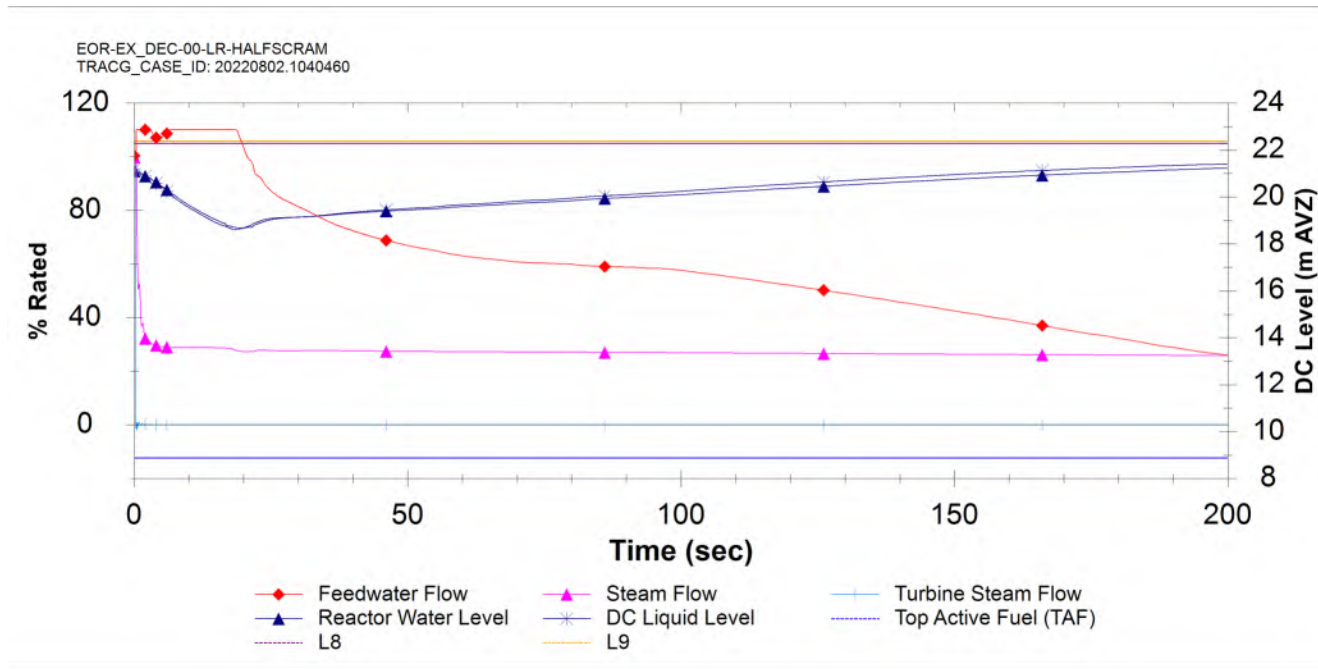


Figure 15.5-132: Complex Sequence Generator Load Rejection (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

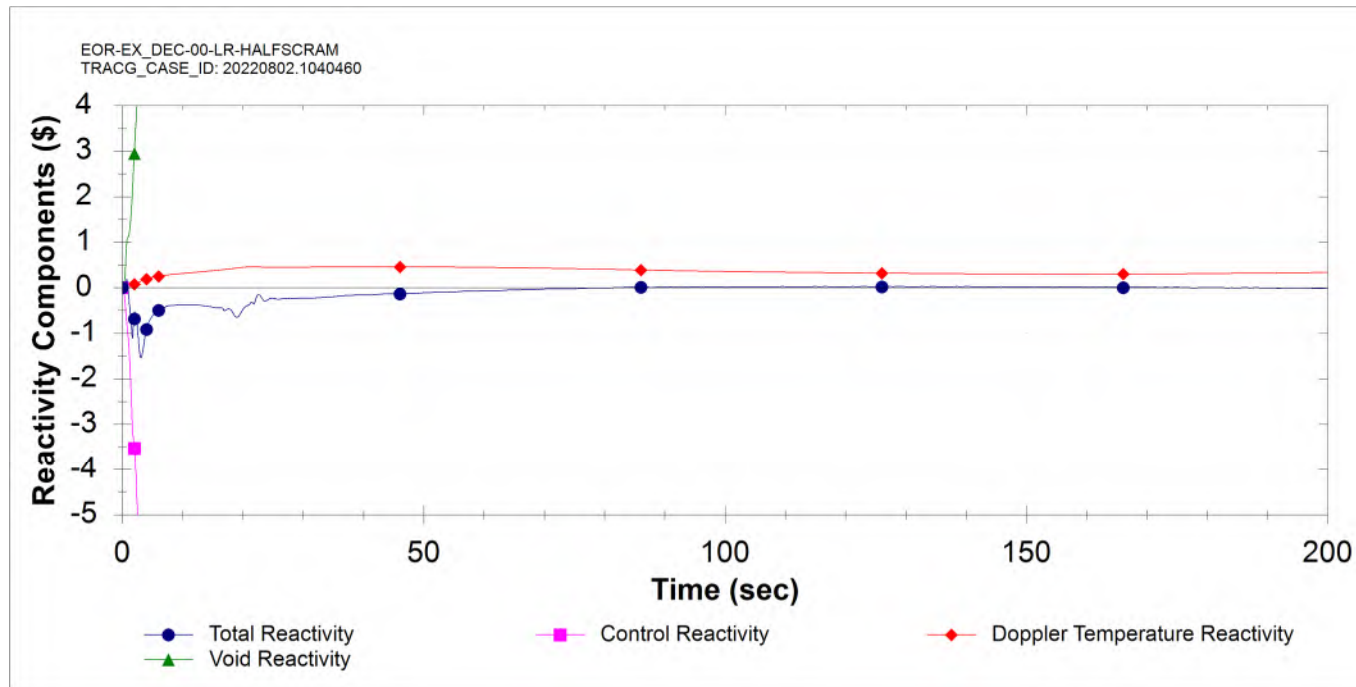


Figure 15.5-133: Complex Sequence Generator Load Rejection (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

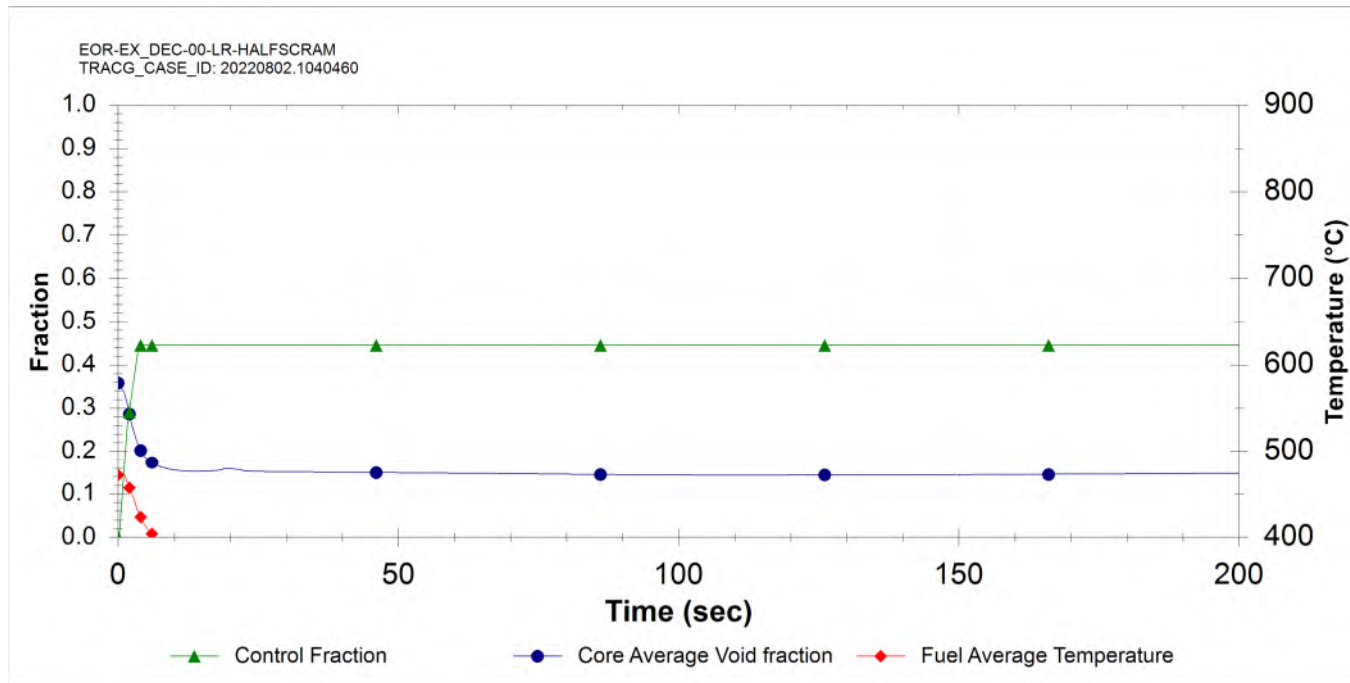


Figure 15.5-134: Complex Sequence Generator Load Rejection (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

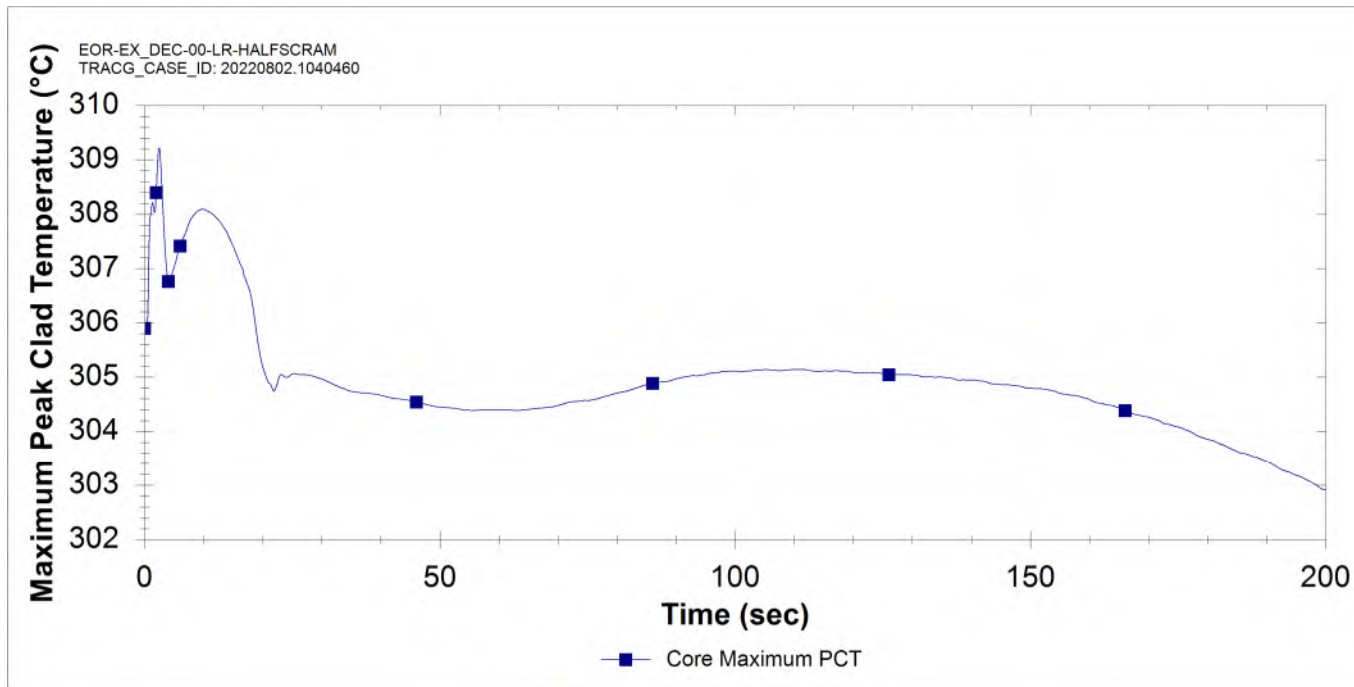


Figure 15.5-135: Complex Sequence Generator Load Rejection (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

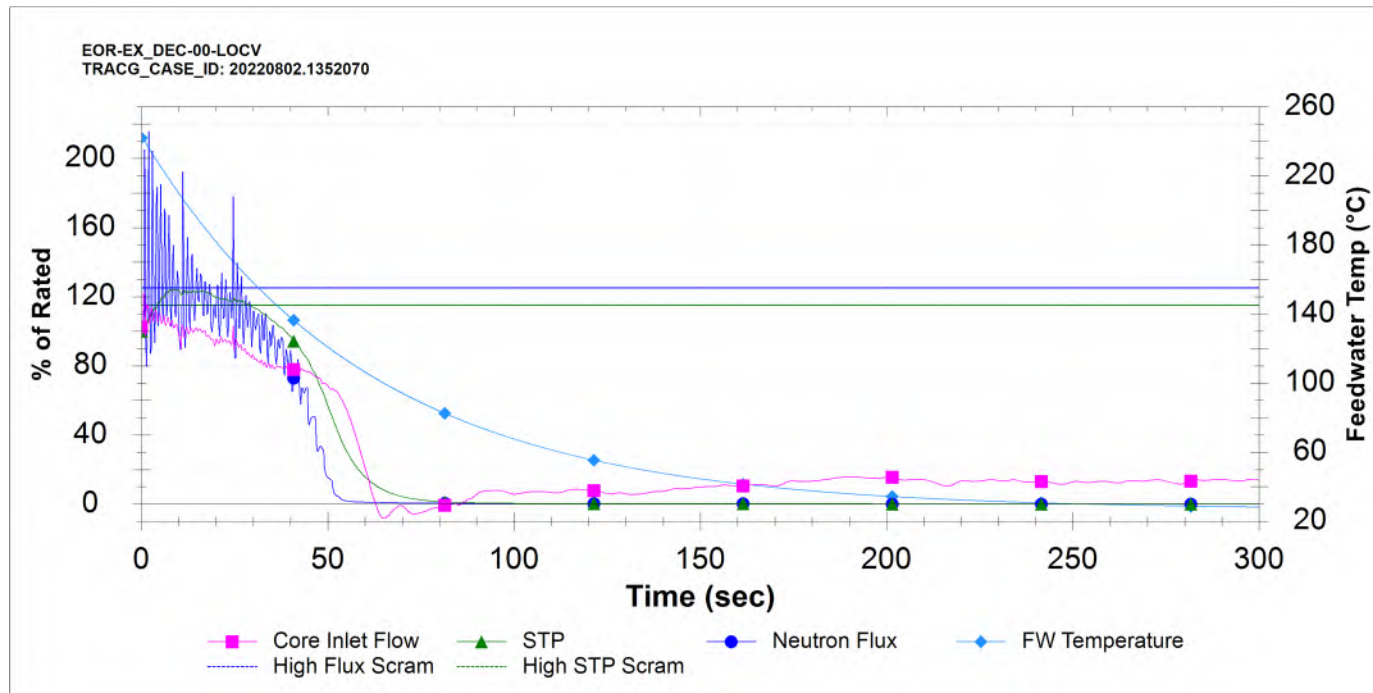


Figure 15.5-136: Loss of Condenser Vacuum (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

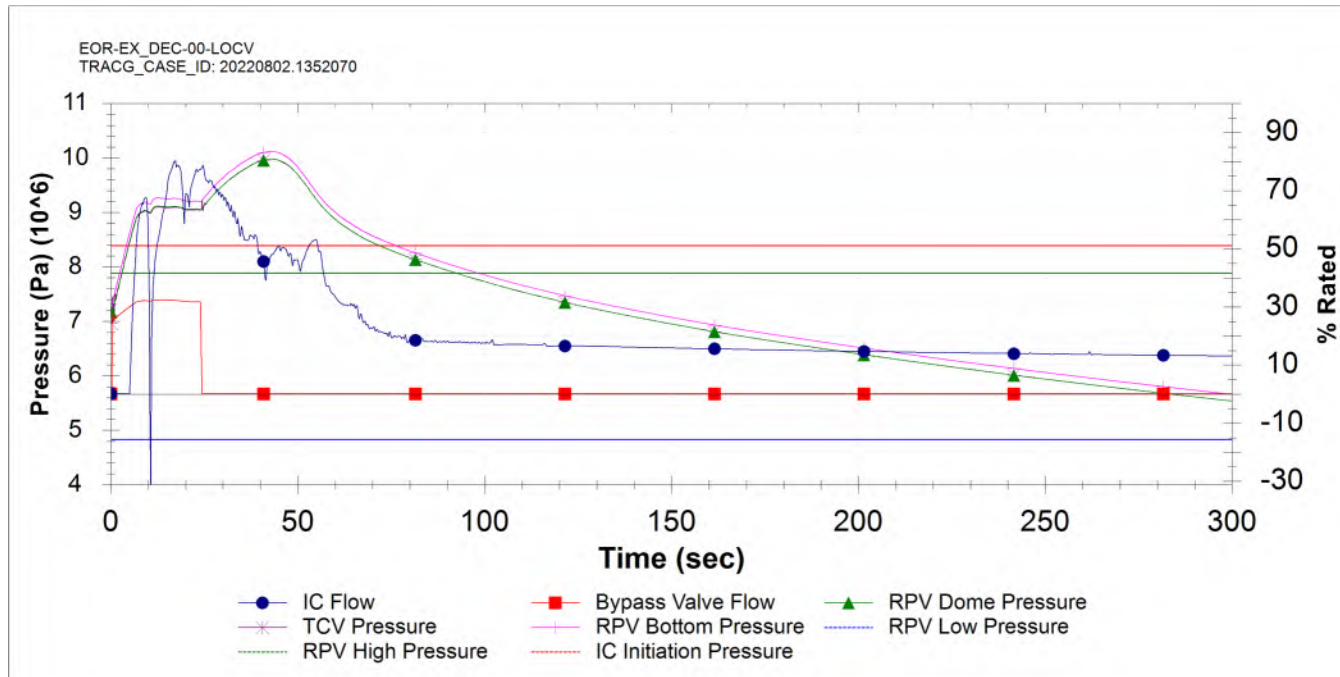


Figure 15.5-137: Loss of Condenser Vacuum (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

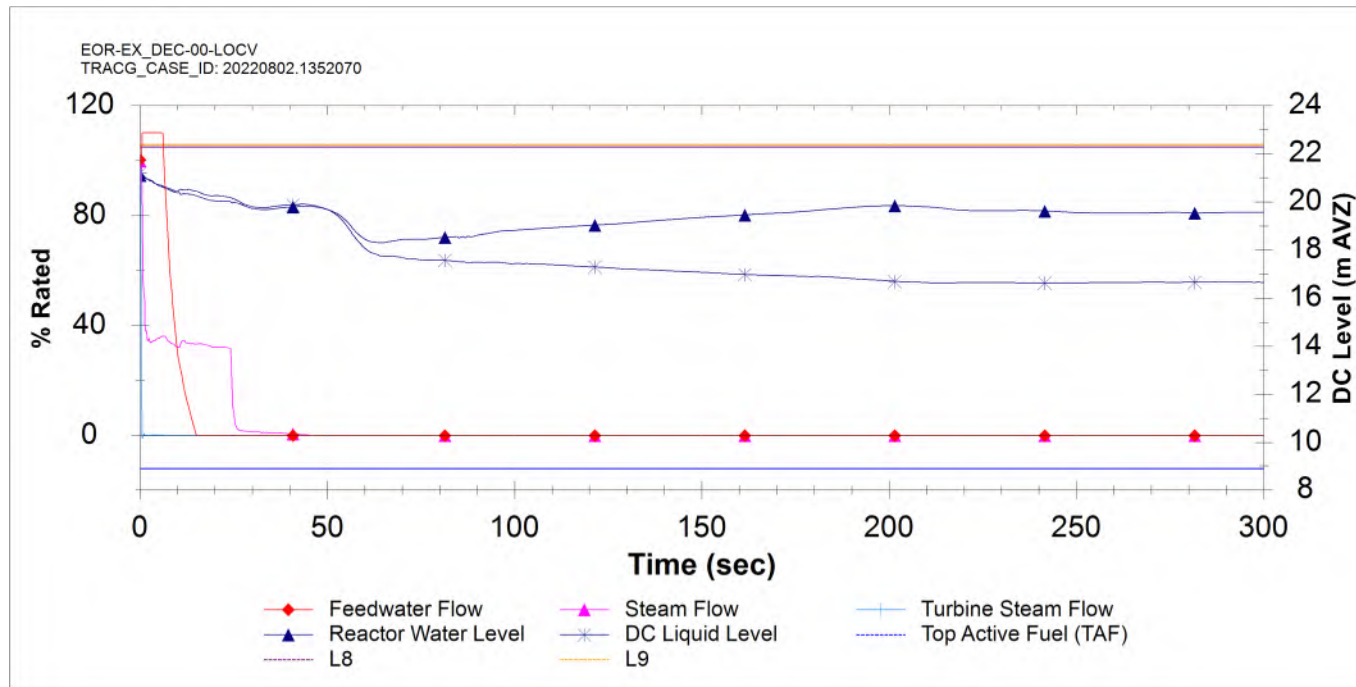


Figure 15.5-138: Loss of Condenser Vacuum (DEC)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

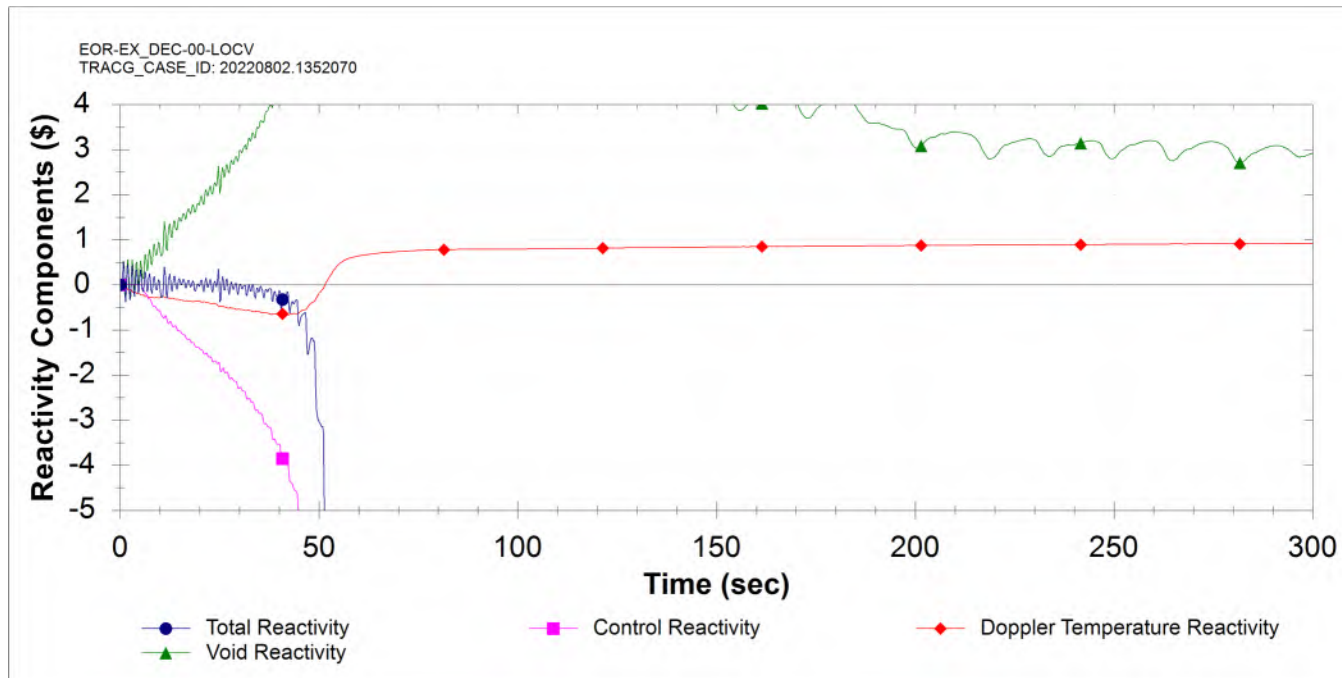


Figure 15.5-139: Loss of Condenser Vacuum (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

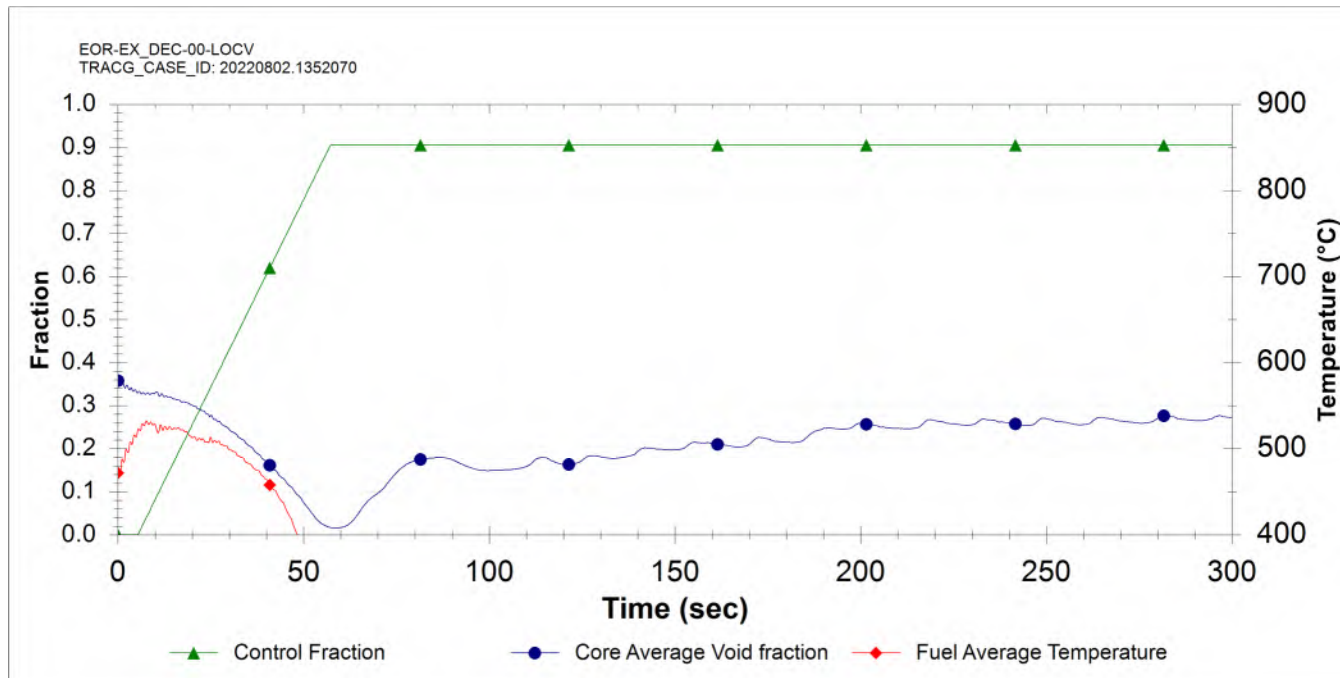


Figure 15.5-140: Loss of Condenser Vacuum (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

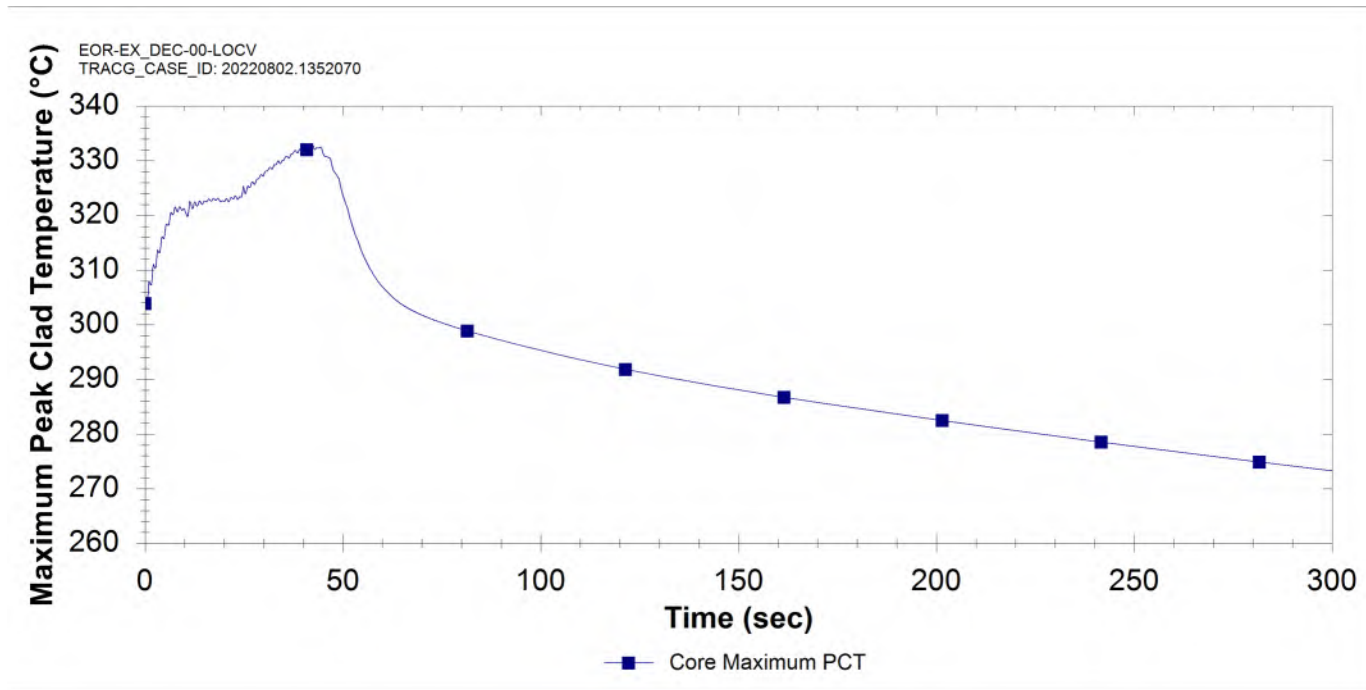


Figure 15.5-141: Loss of Condenser Vacuum (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

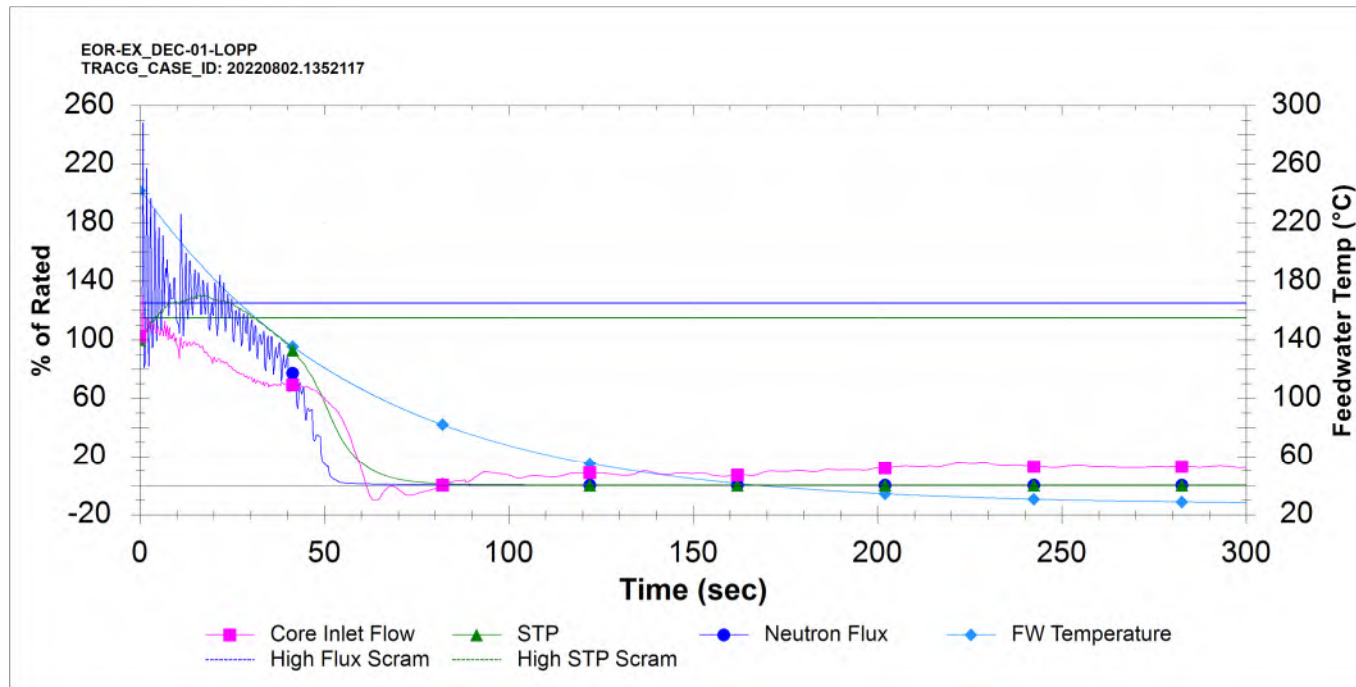


Figure 15.5-142: Loss of Preferred Power (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

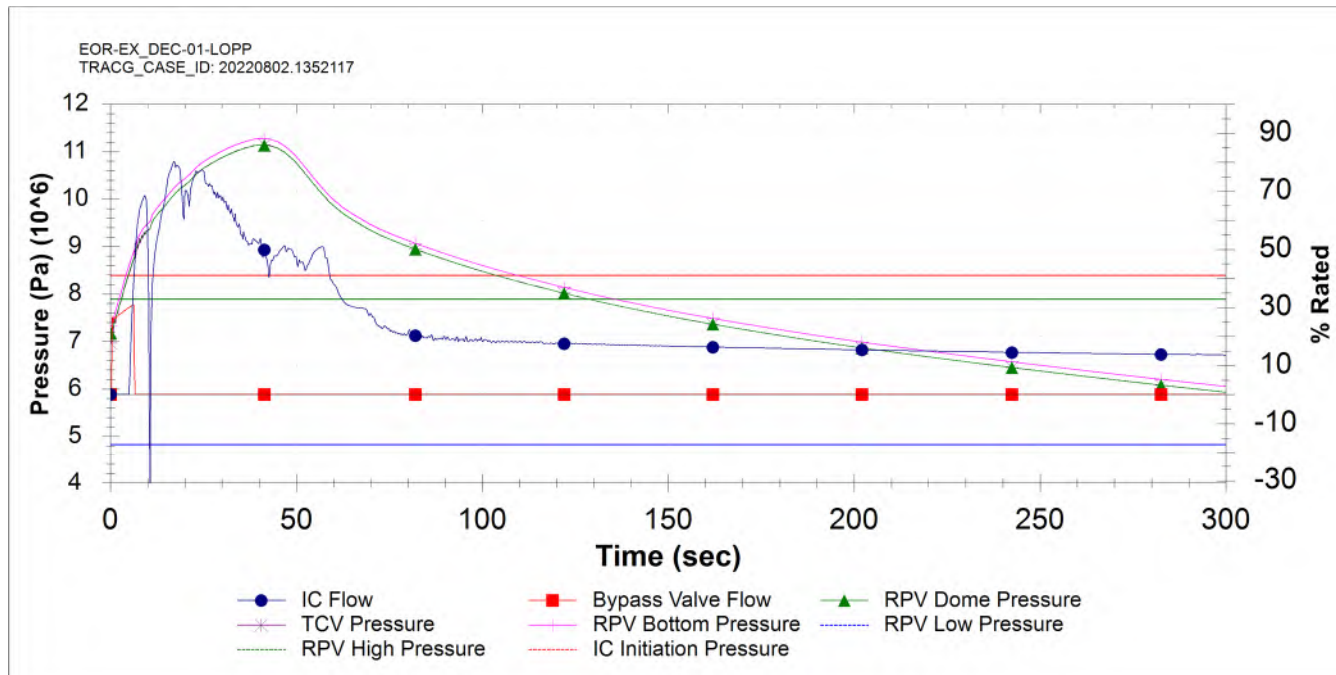


Figure 15.5-143: Loss of Preferred Power (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

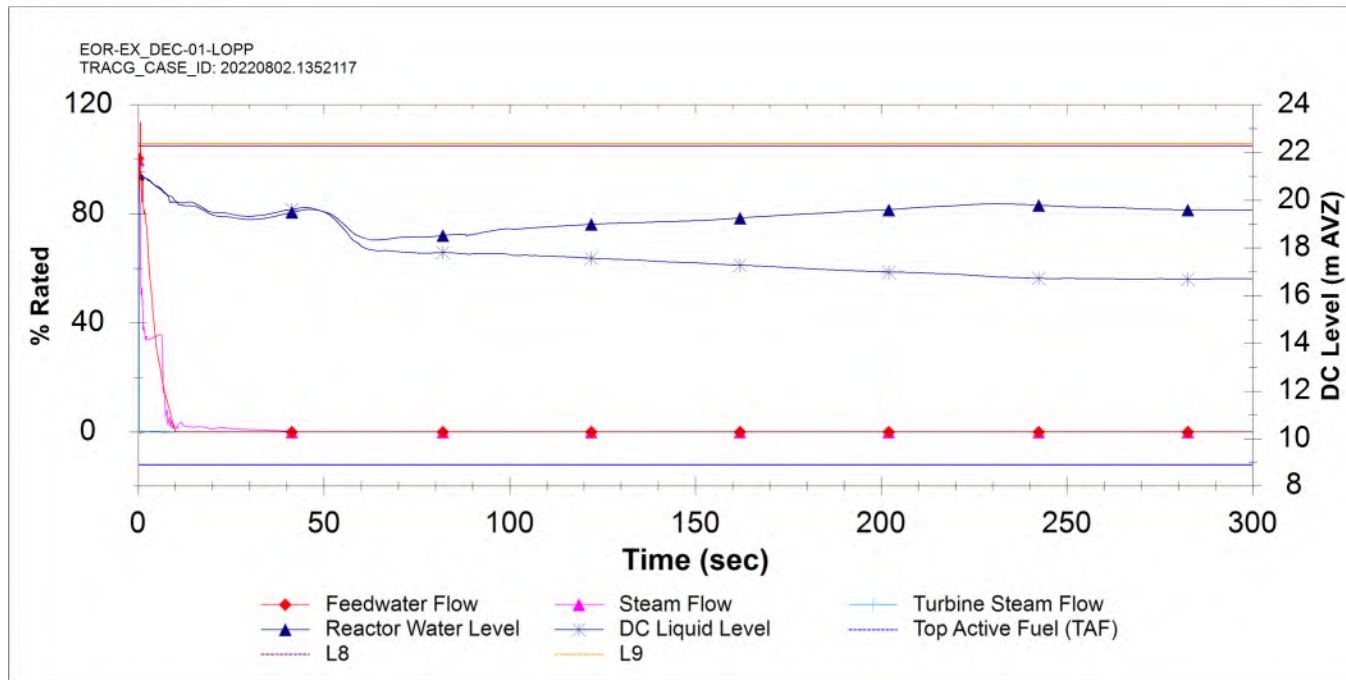


Figure 15.5-144: Loss of Preferred Power (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

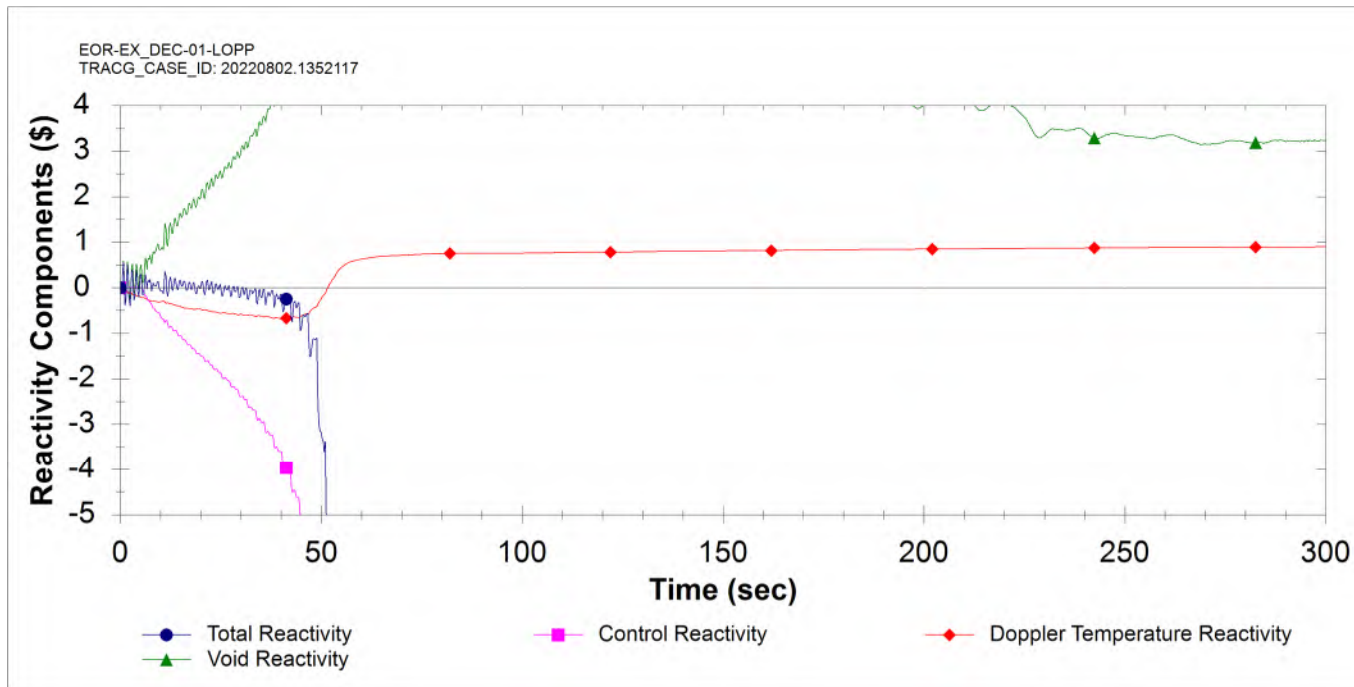


Figure 15.5-145: Loss of Preferred Power (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

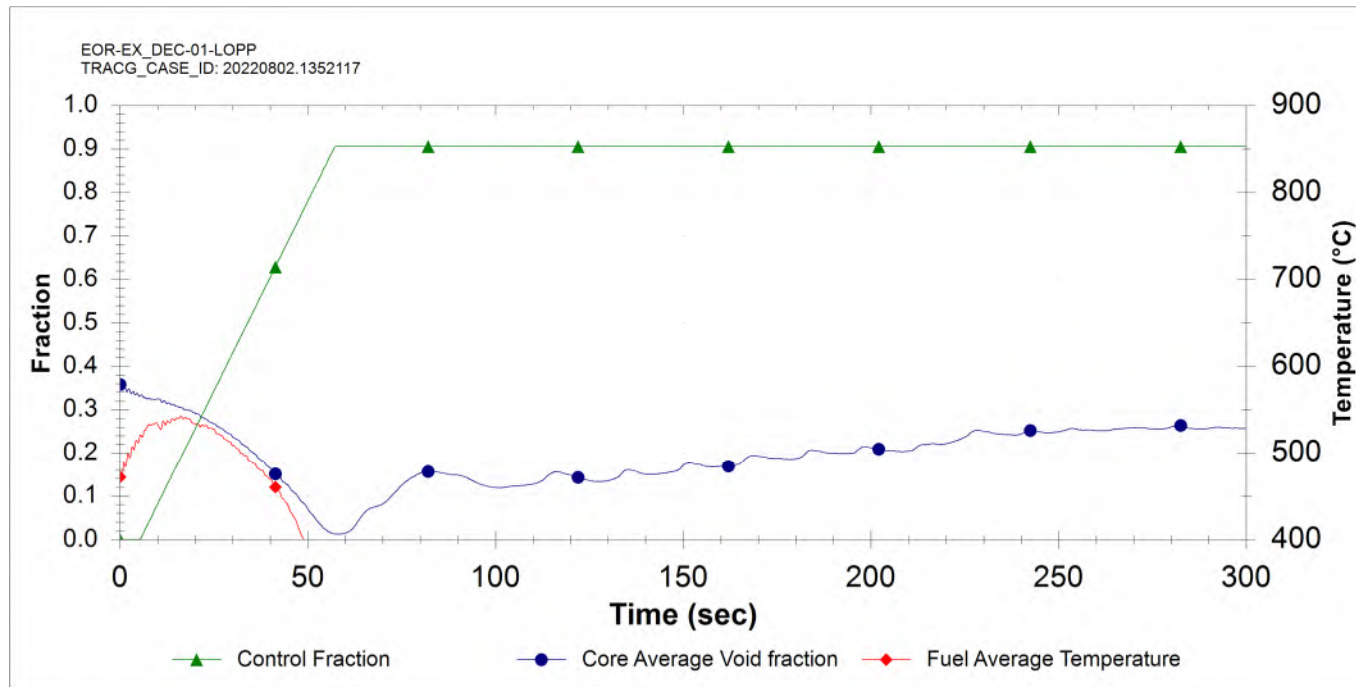


Figure 15.5-146: Loss of Preferred Power (DEC)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

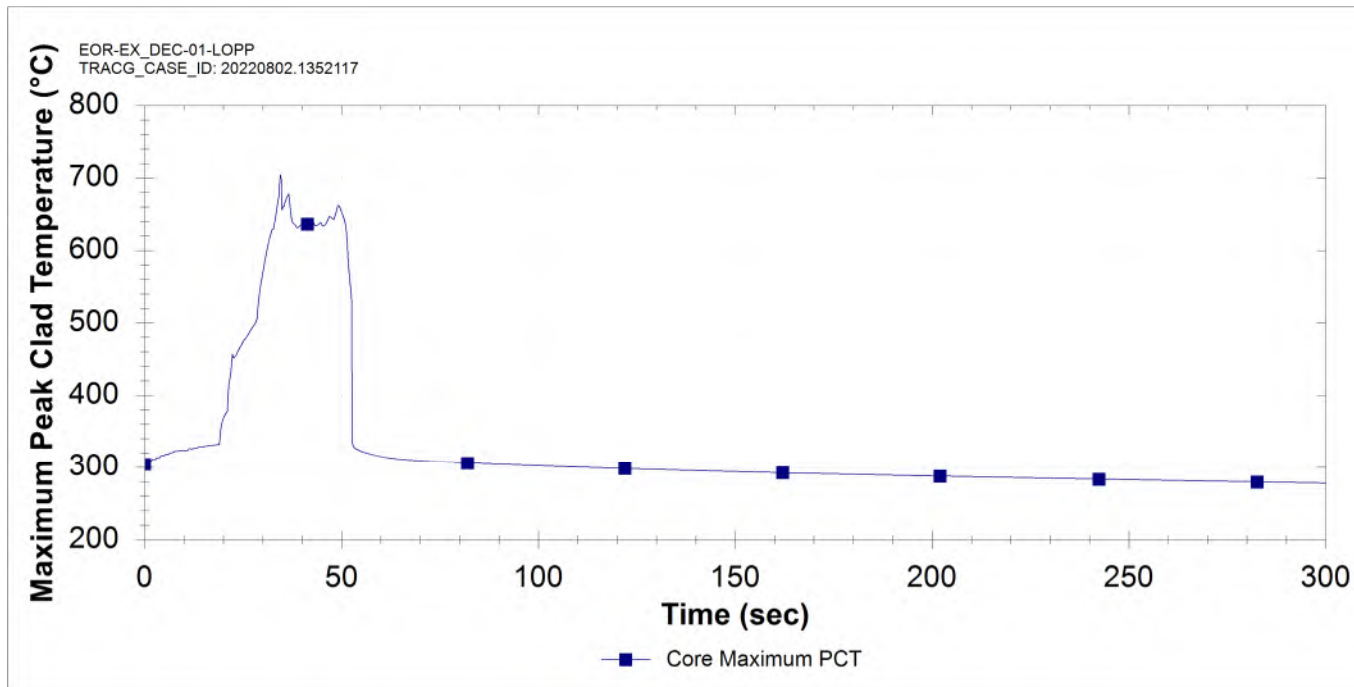
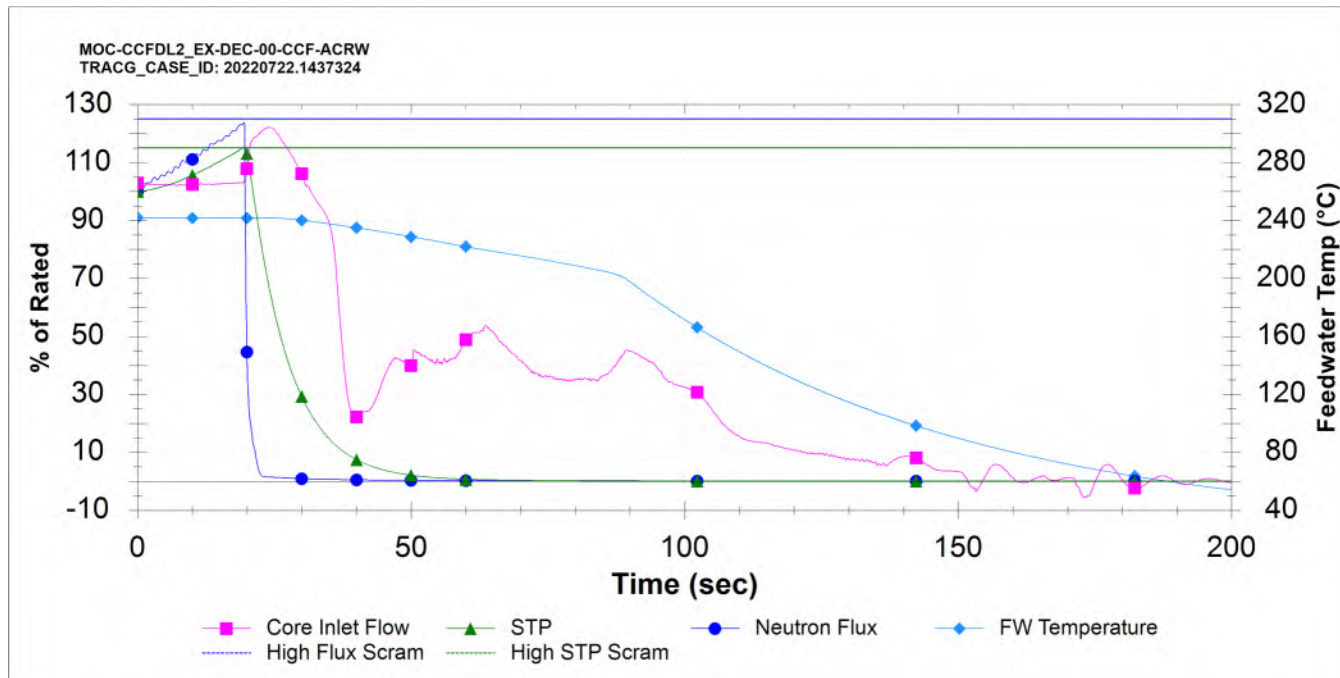


Figure 15.5-147: Loss of Preferred Power (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-148: All Control Rod Withdrawal at Power (ACRW)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

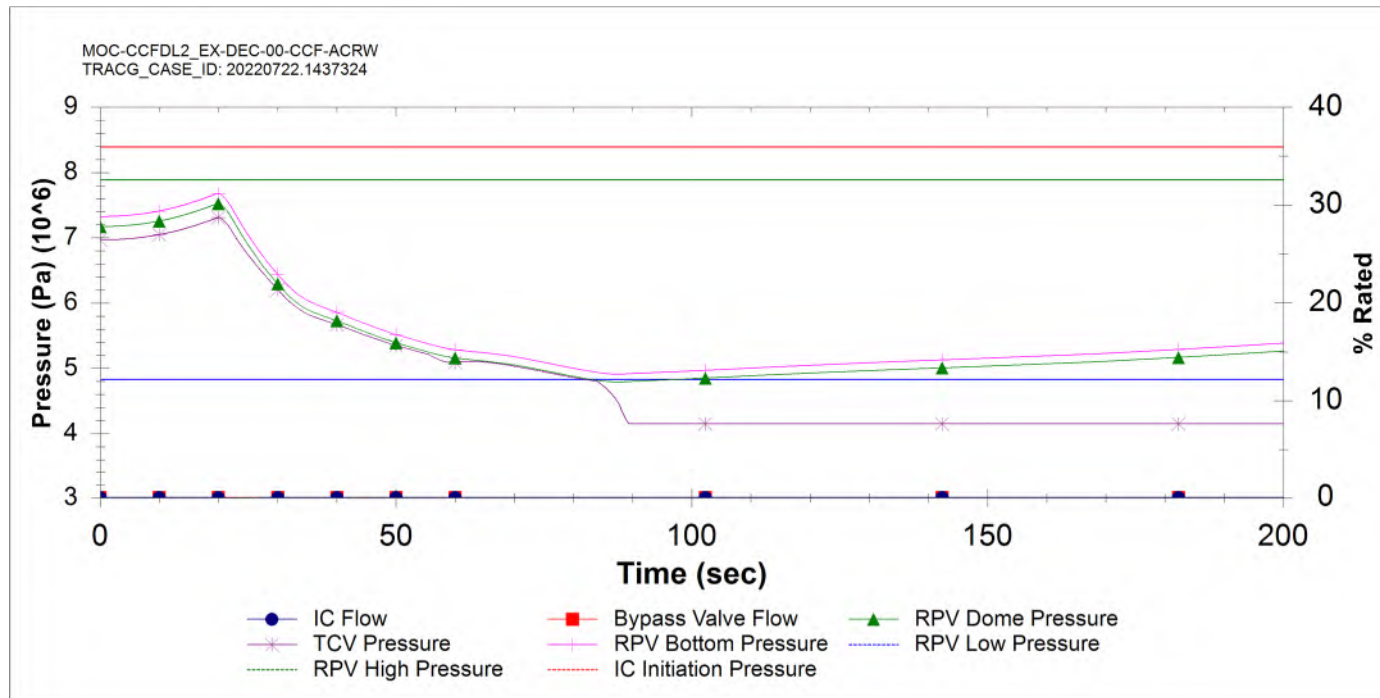
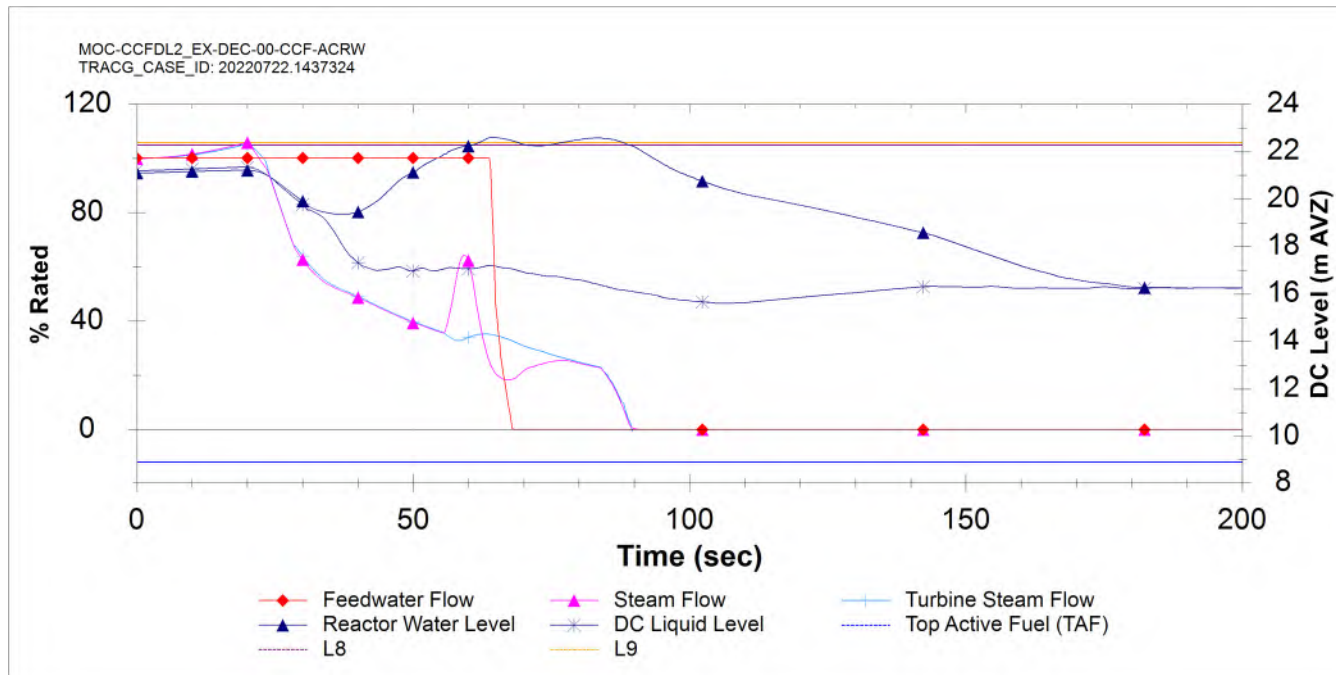


Figure 15.5-149: All Control Rod Withdrawal at Power (ACRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-150: All Control Rod Withdrawal at Power (ACRW)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

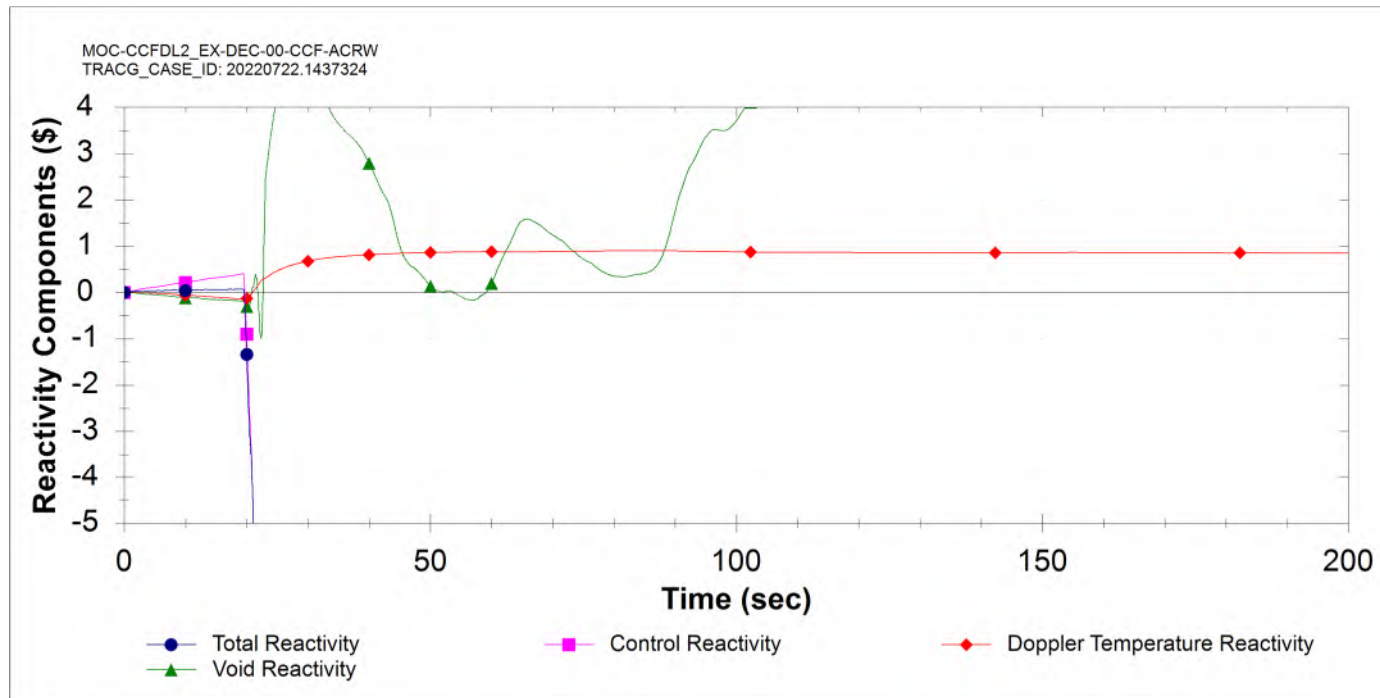
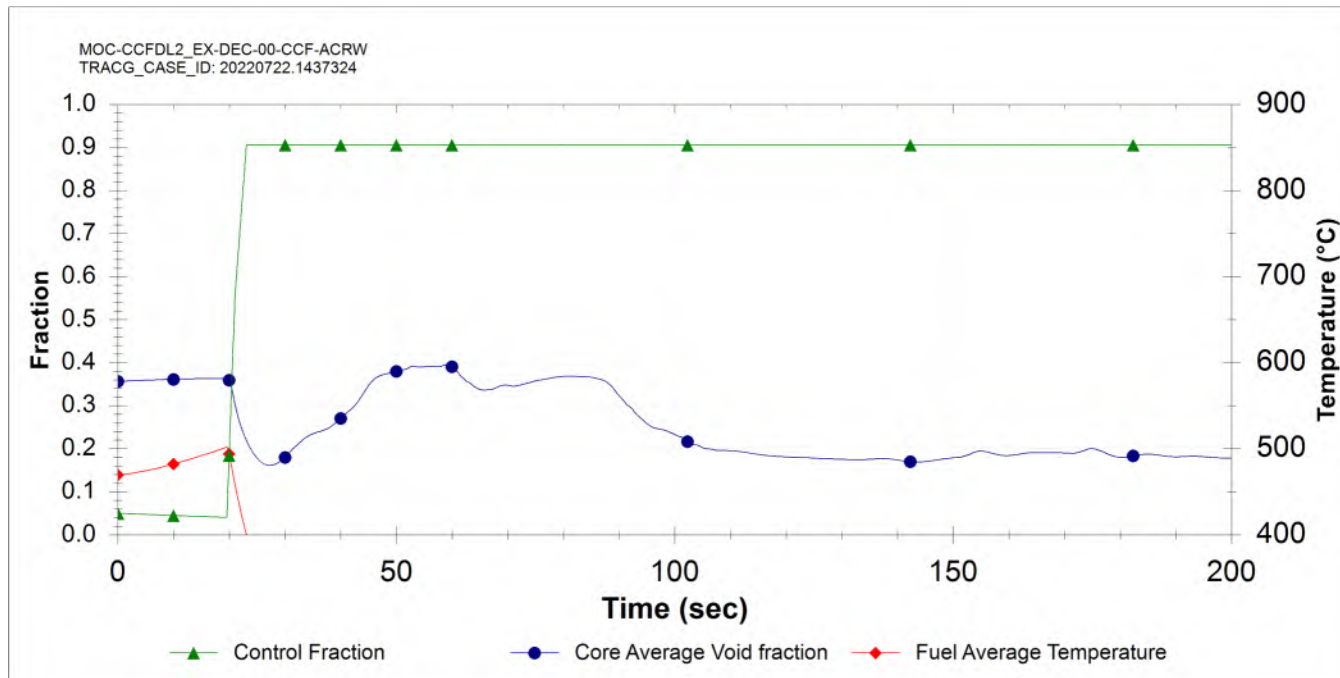


Figure 15.5-151: All Control Rod Withdrawal at Power (ACRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-152: All Control Rod Withdrawal at Power (ACRW)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

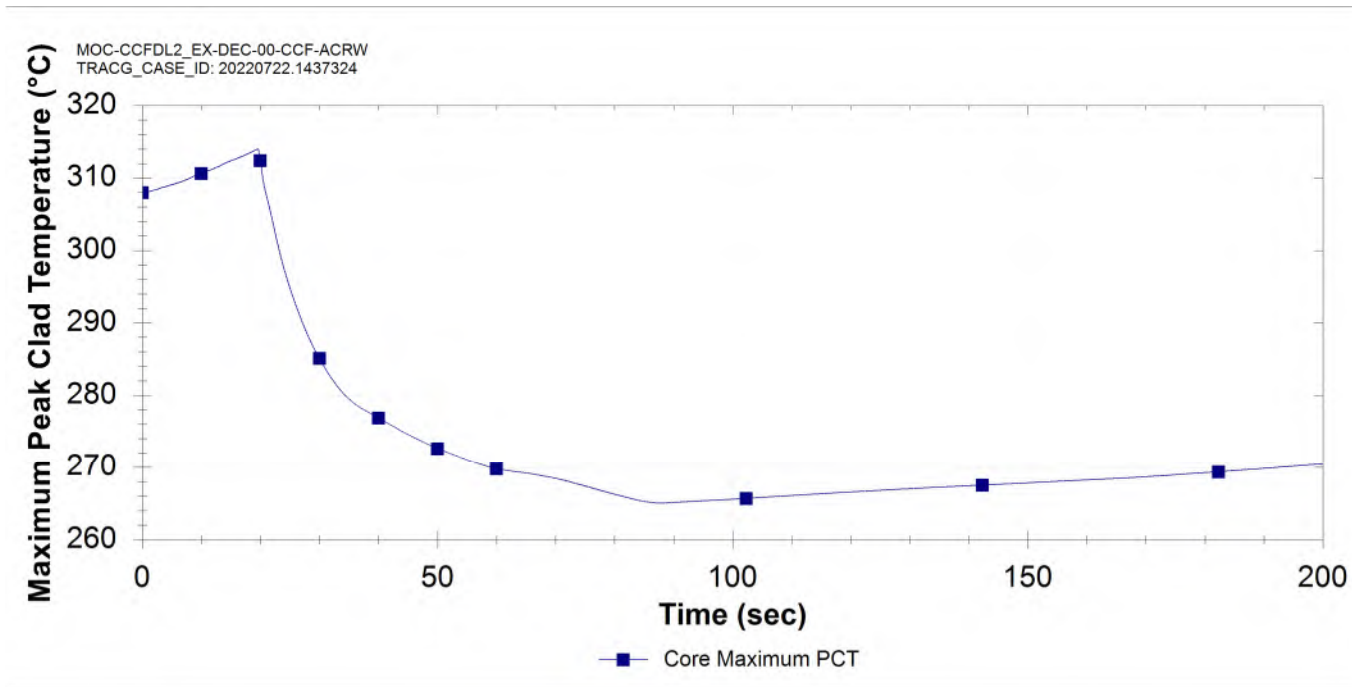


Figure 15.5-153: All Control Rod Withdrawal at Power (ACRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

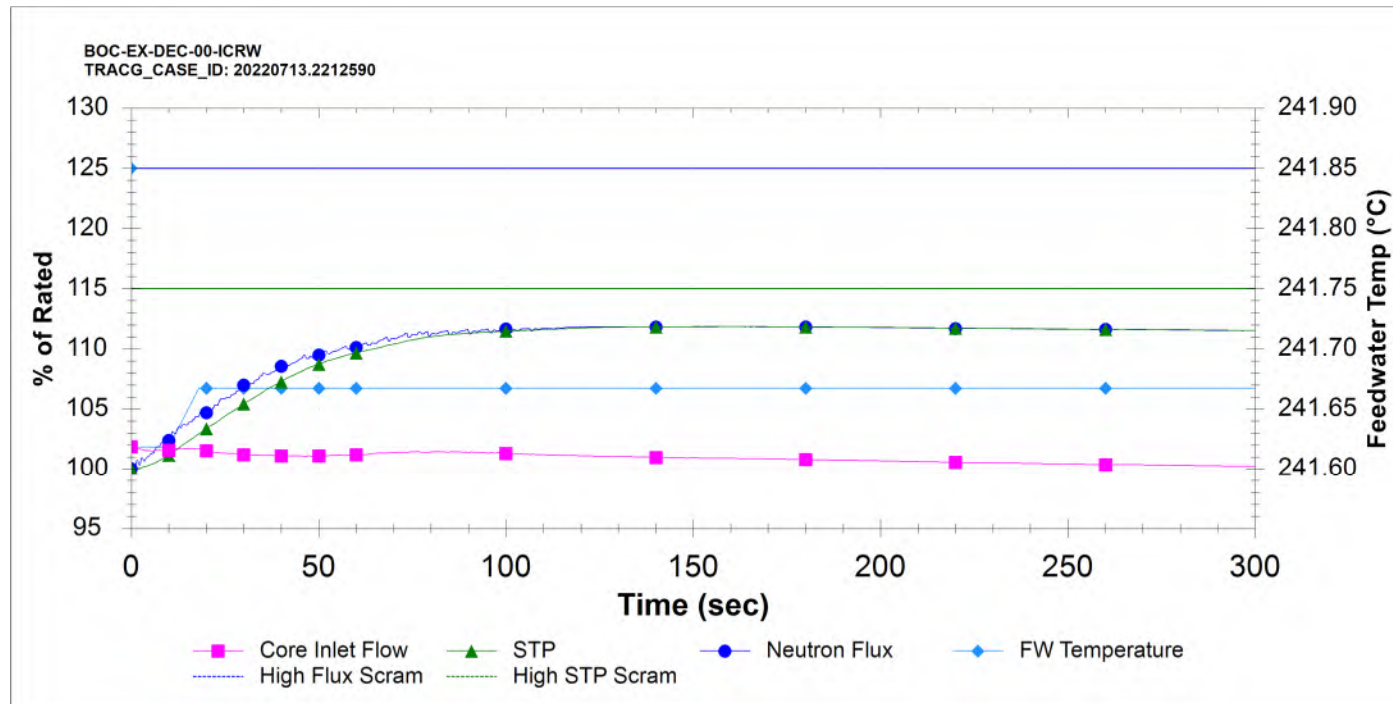
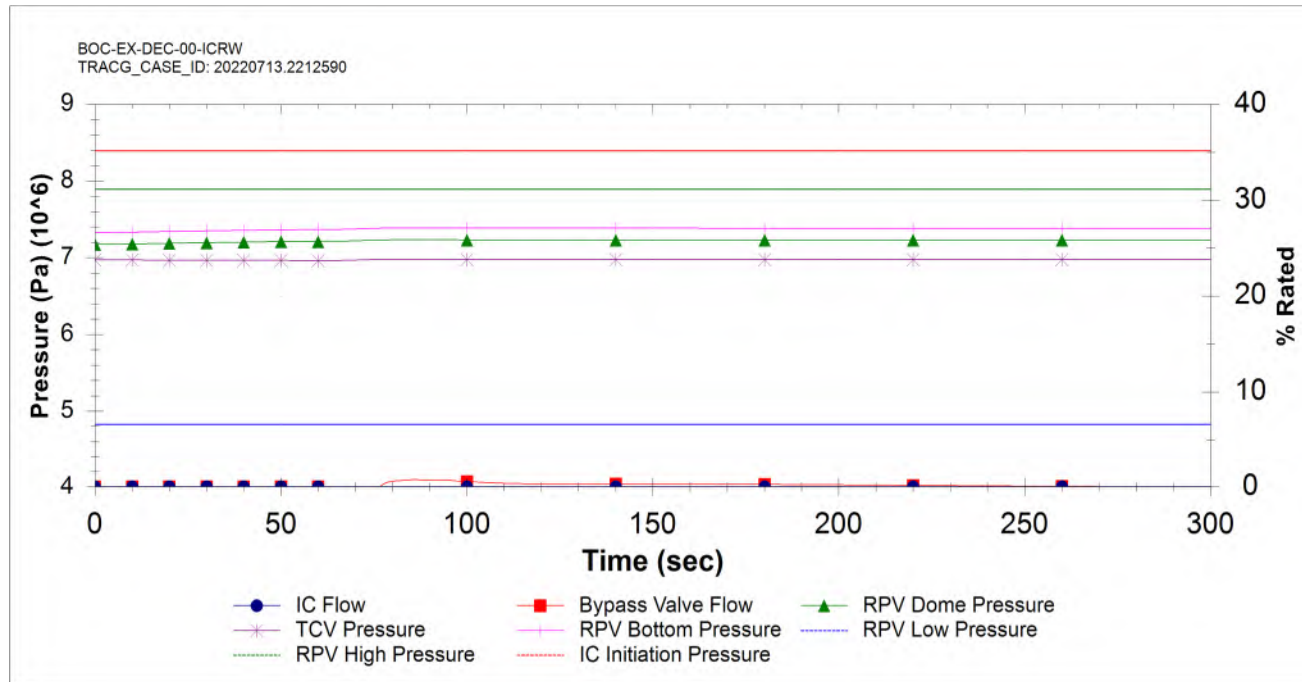


Figure 15.5-154: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

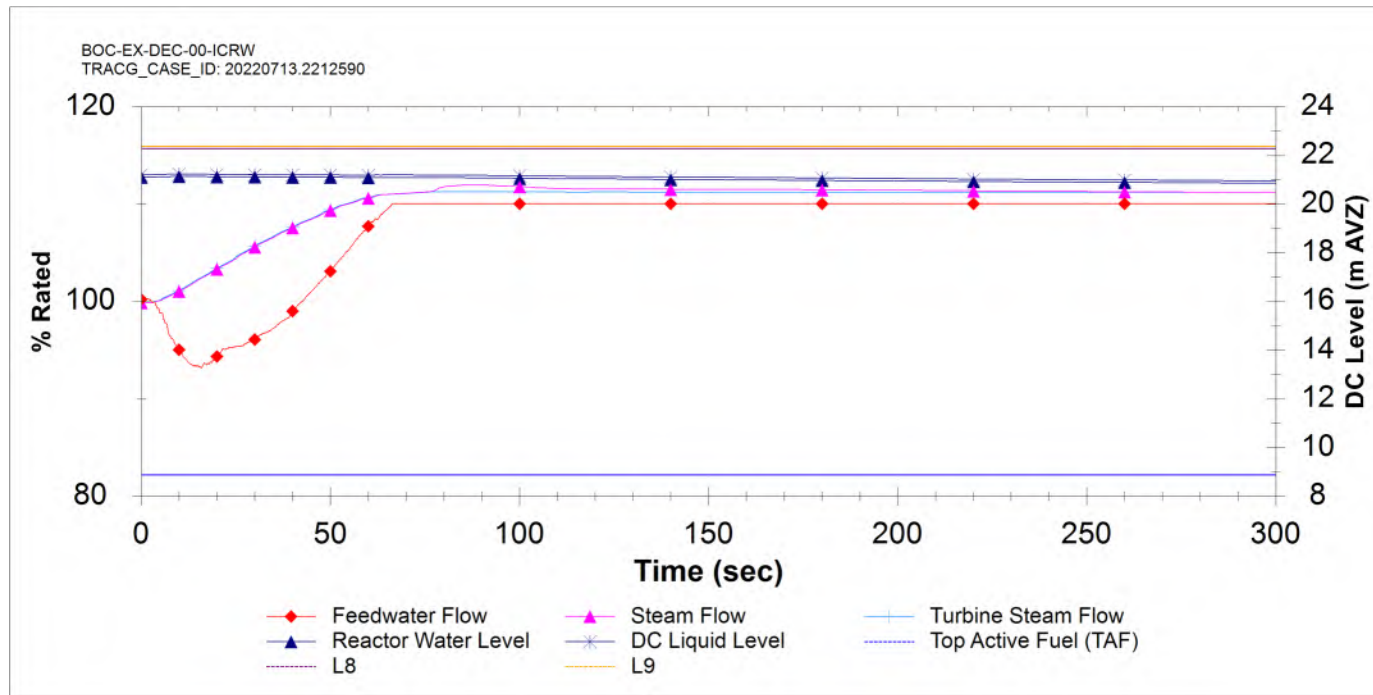


NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-155: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 15.5-156: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)**

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

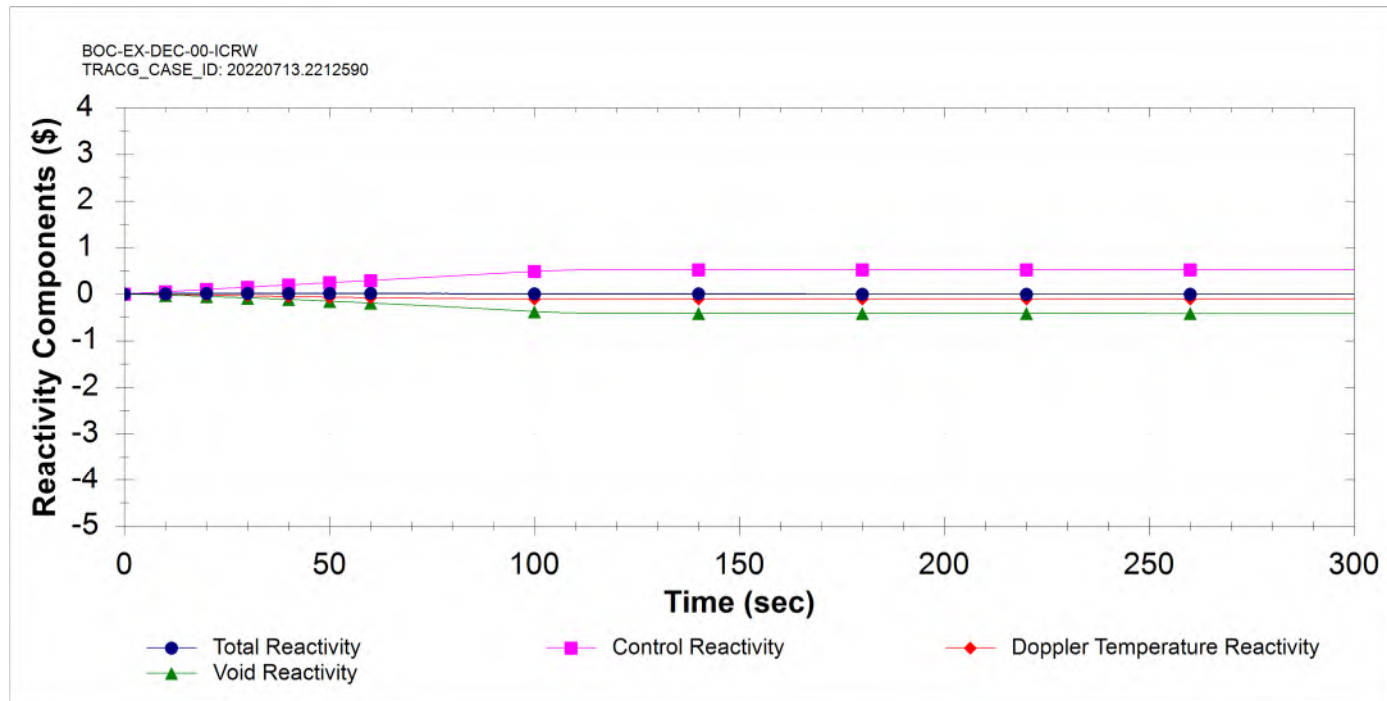


Figure 15.5-157: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

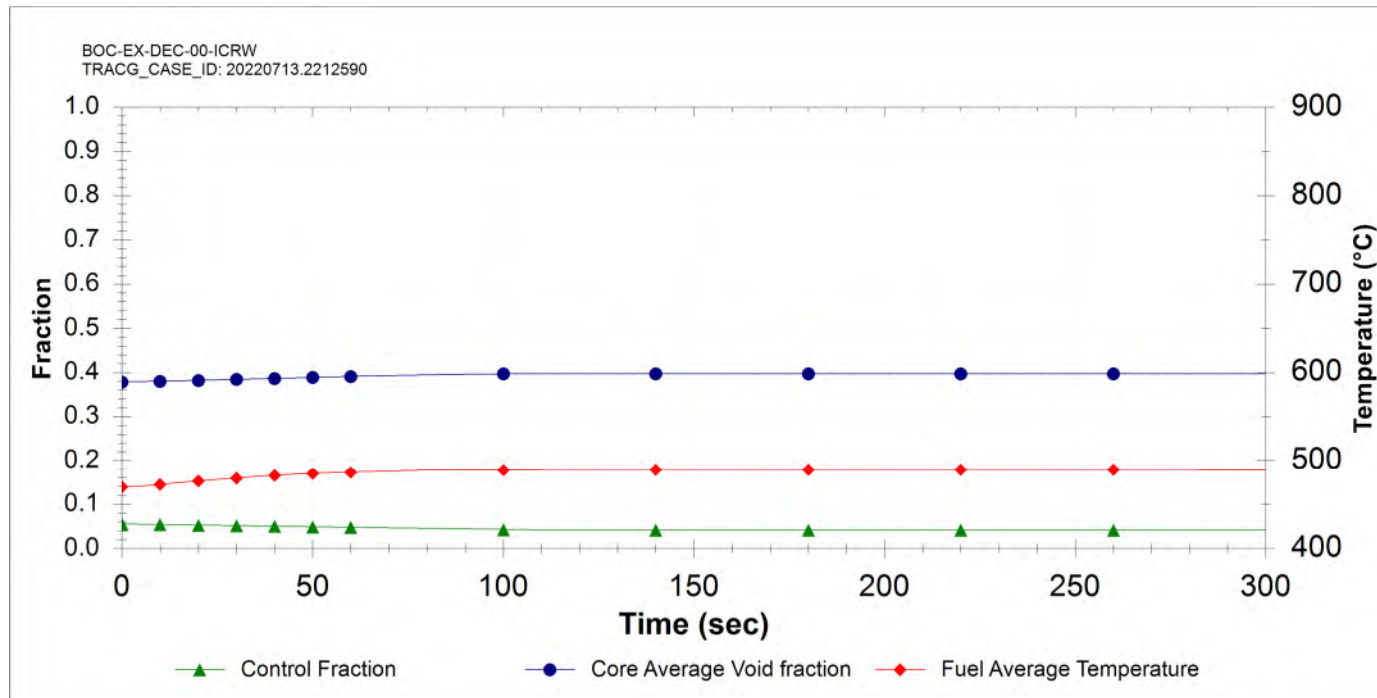


Figure 15.5-158: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

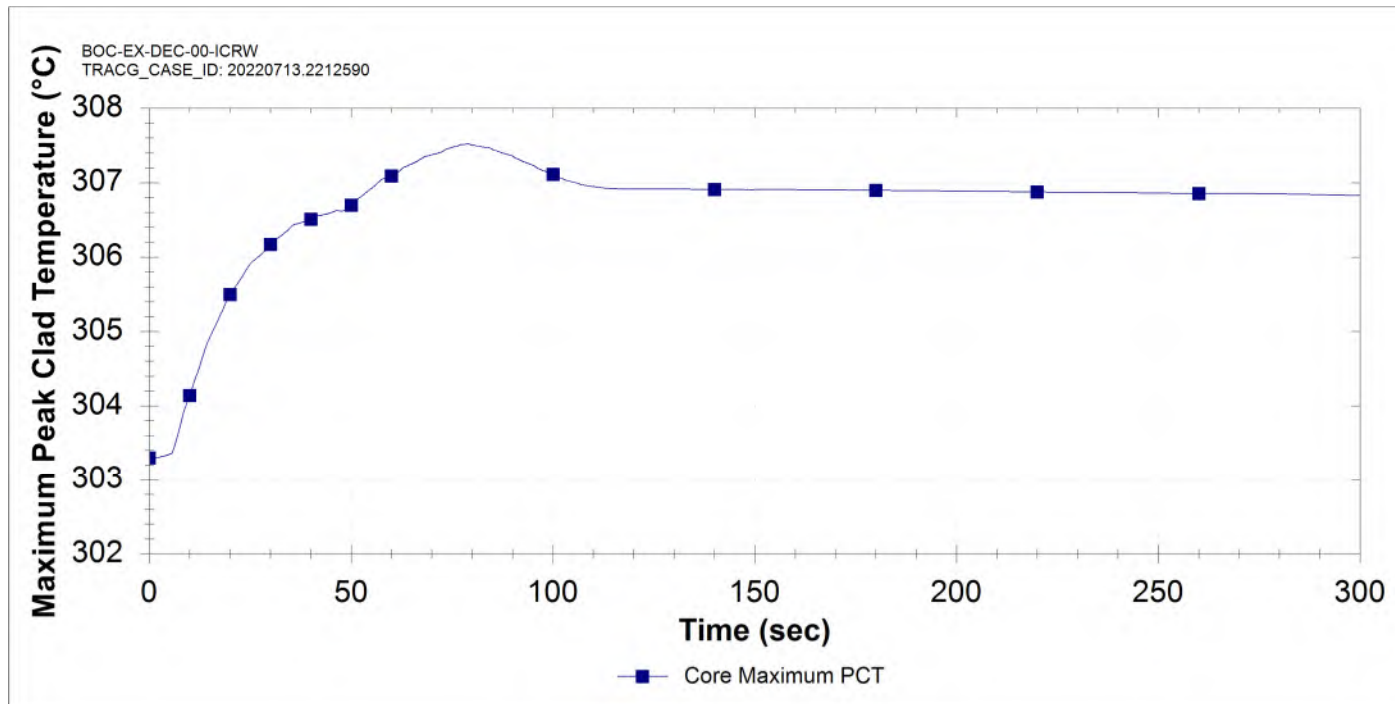


Figure 15.5-159: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

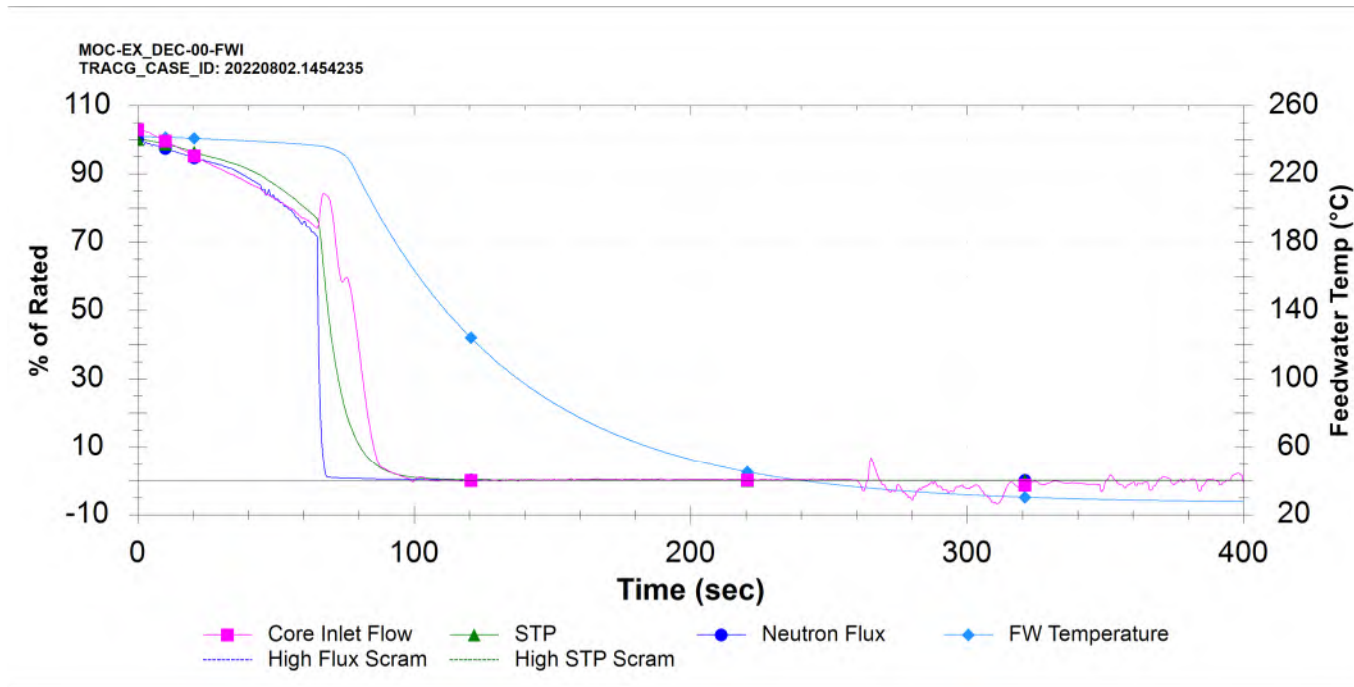


Figure 15.5-160: Feedwater Isolation (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

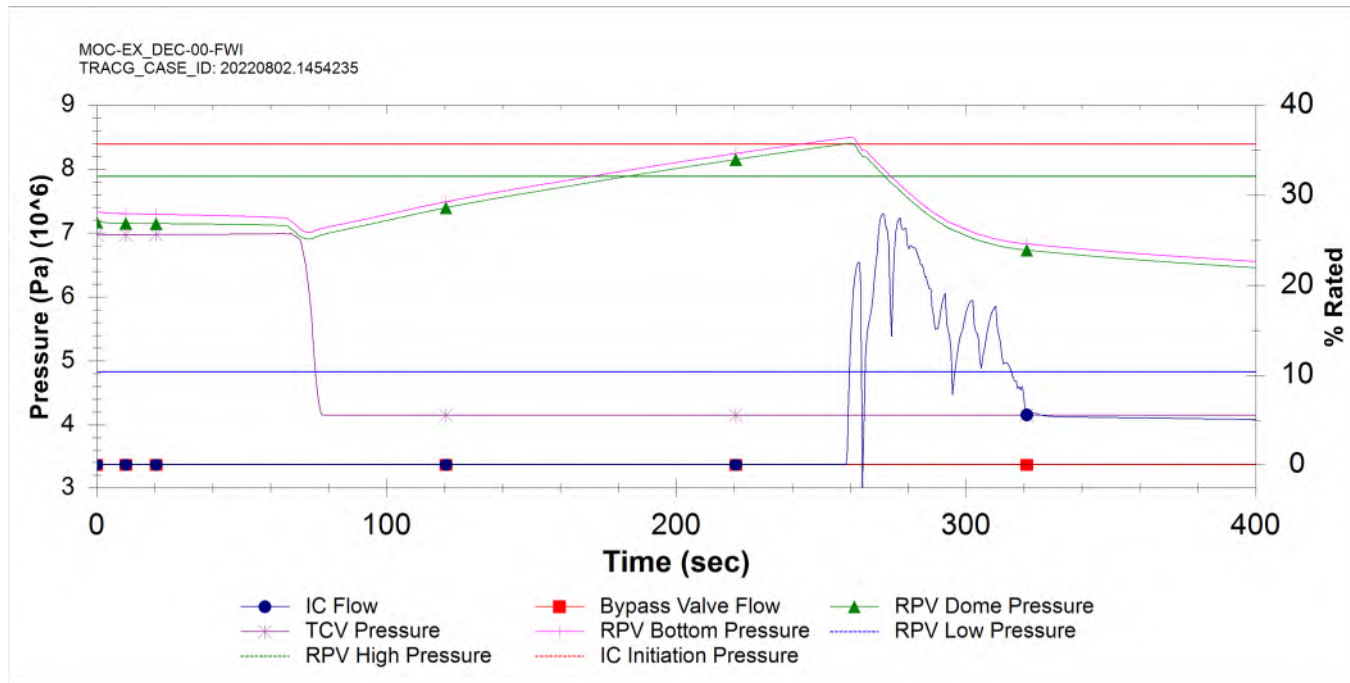


Figure 15.5-161: Feedwater Isolation (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

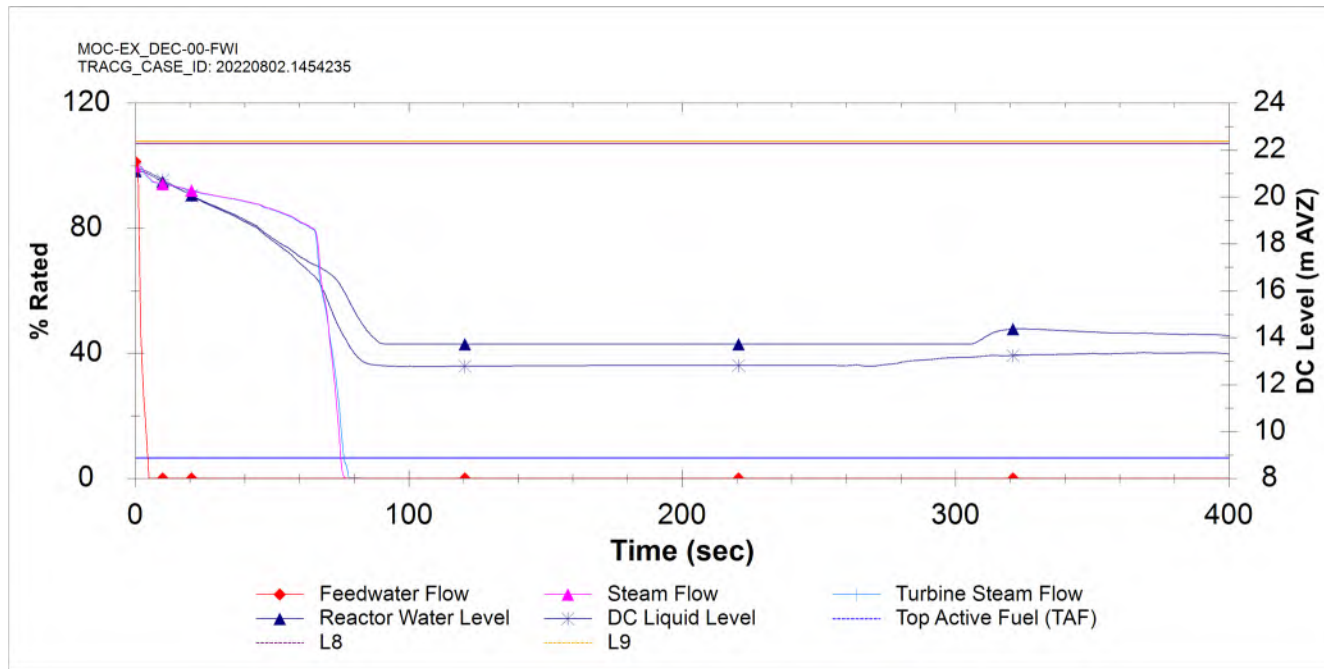


Figure 15.5-162: Feedwater Isolation (DEC)



NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

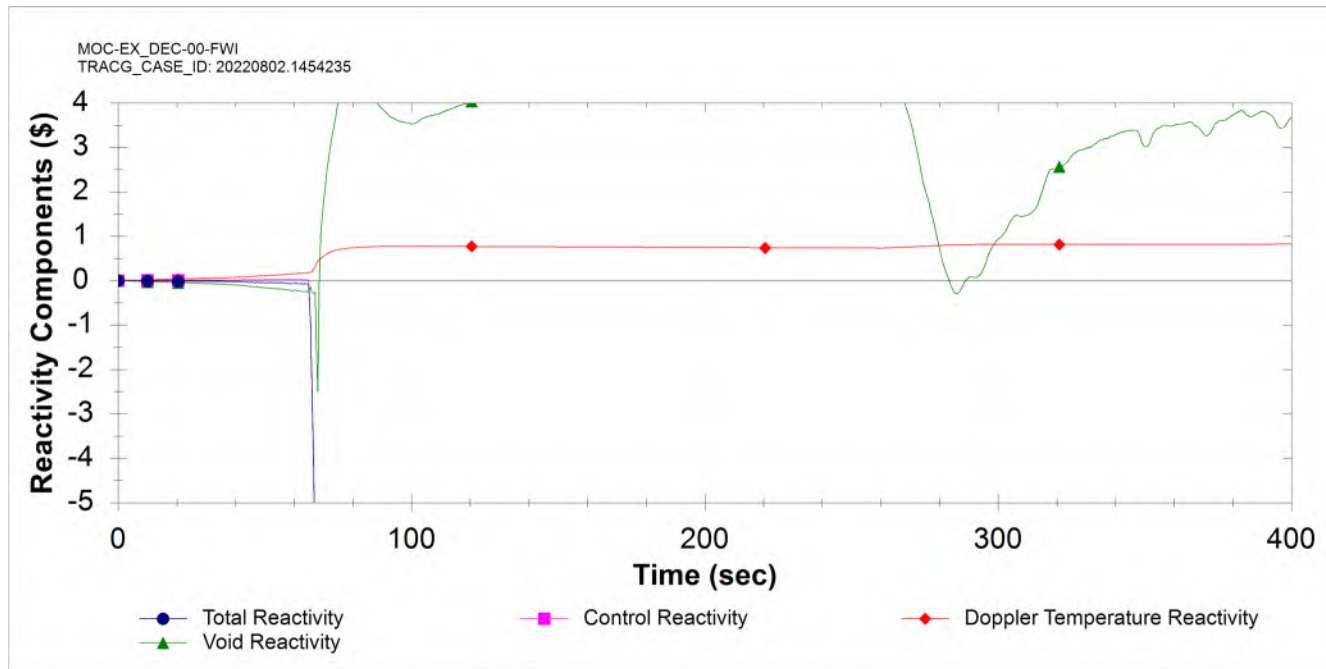


Figure 15.5-163: Feedwater Isolation (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

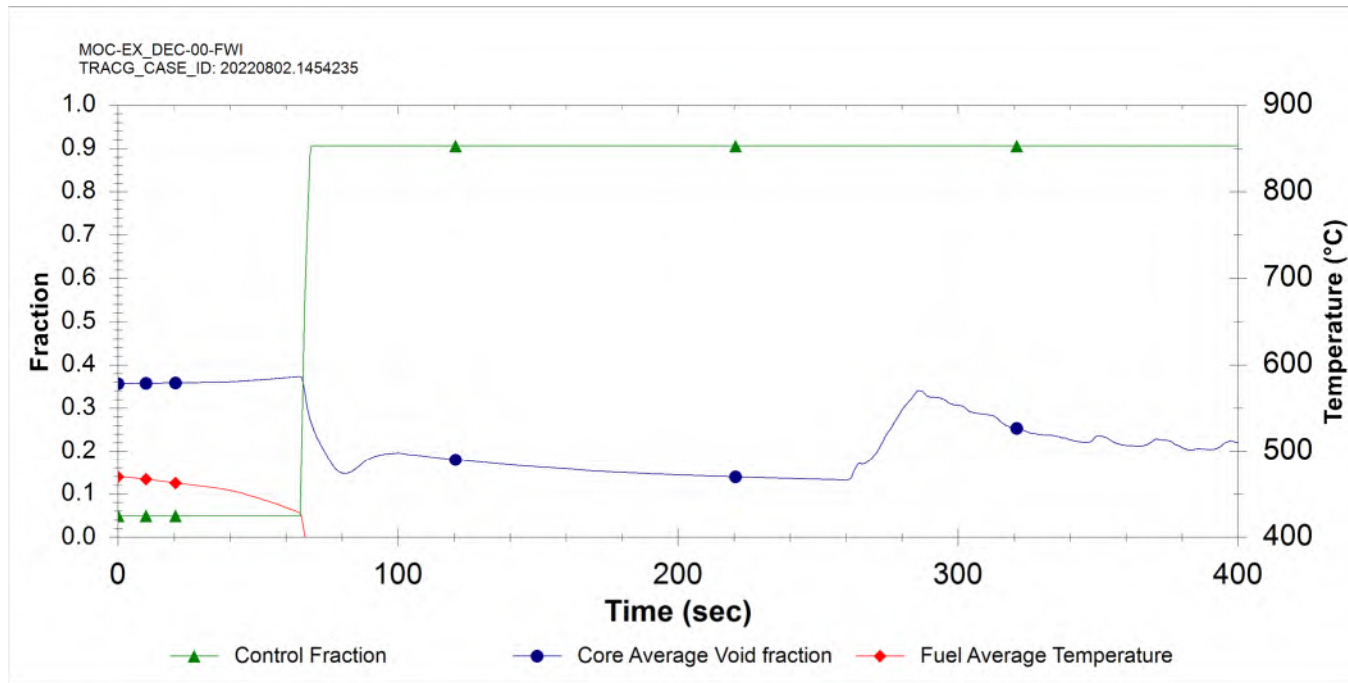


Figure 15.5-164: Feedwater Isolation (DEC)

NEDO-33965 REVISION 0  
NON-PROPRIETARY INFORMATION

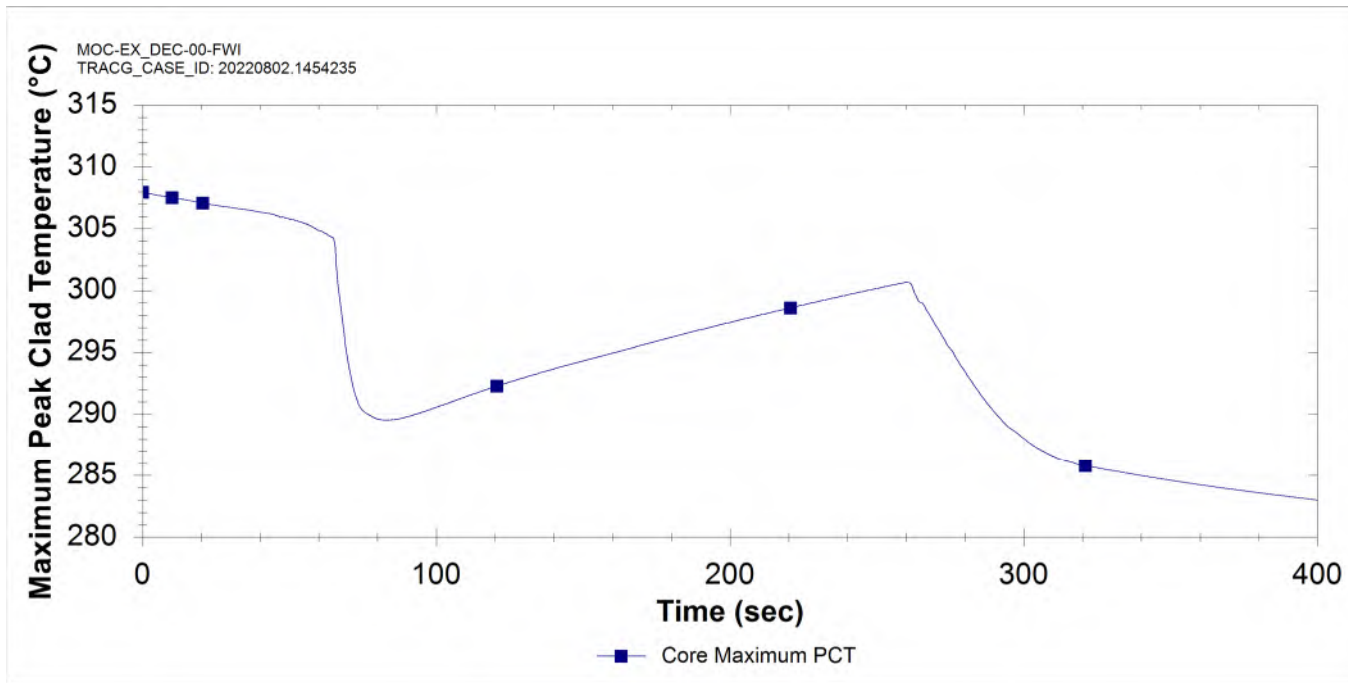


Figure 15.5-165: Feedwater Isolation (DEC)

Figure 15.5-166: Not Used

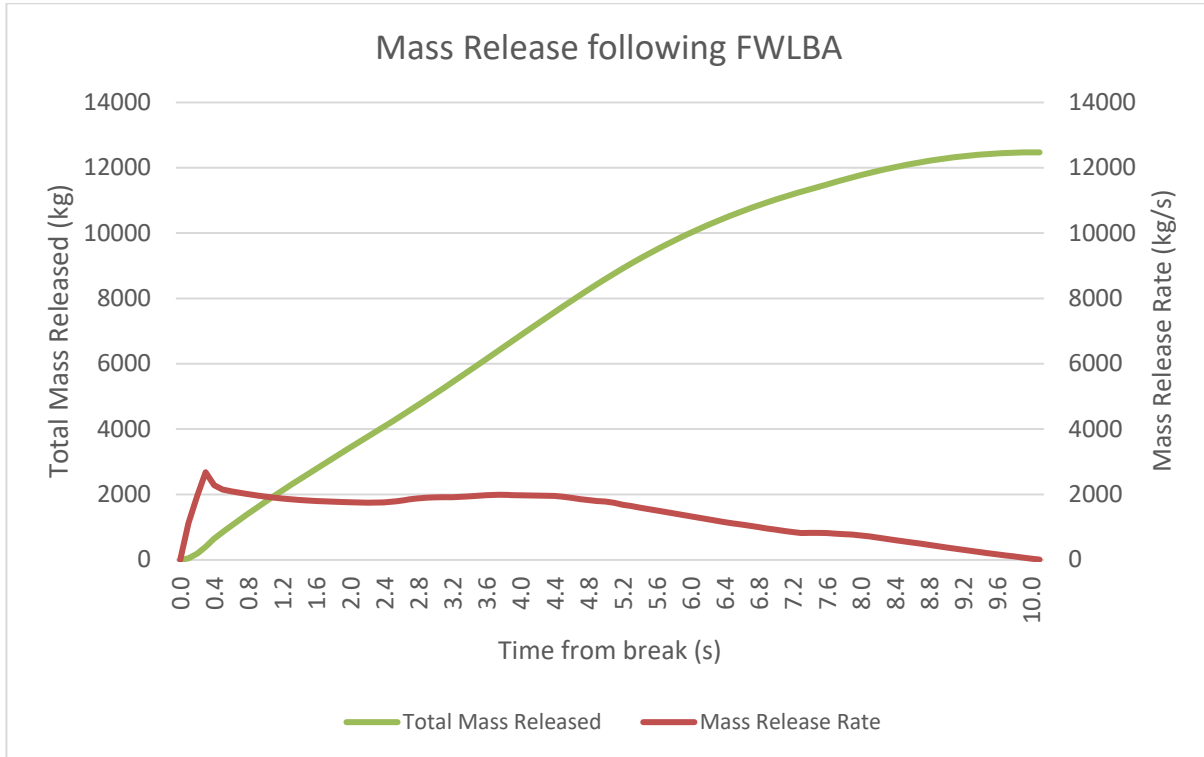
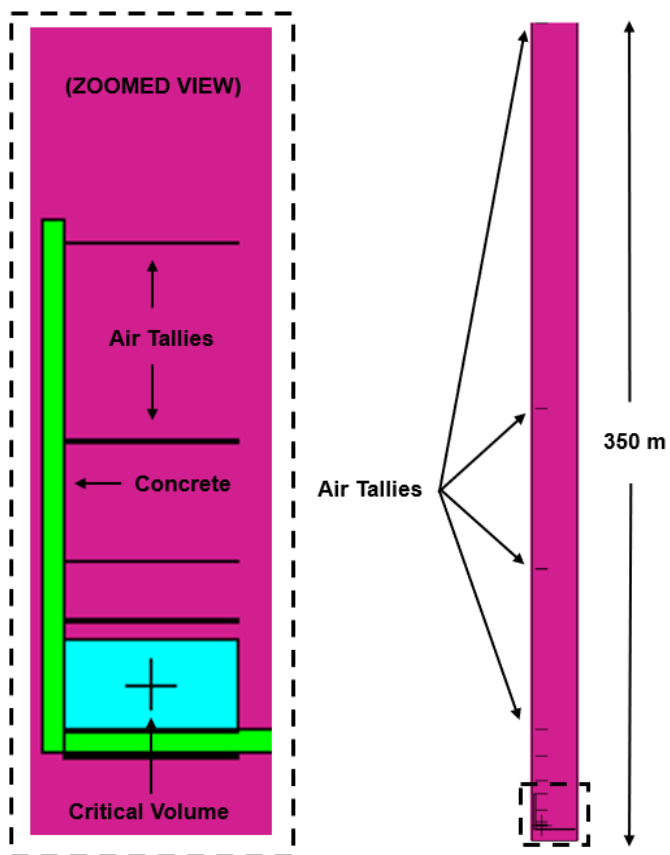


Figure 15.5-167: FWLB Dose Analysis Mass Release vs Time



**Figure 15.5-168: MCNP-06P Problem Geometry for Out of Core Criticality Model**

Note: Dashed lines used to indicate left diagram is zoomed-in version of right diagram.

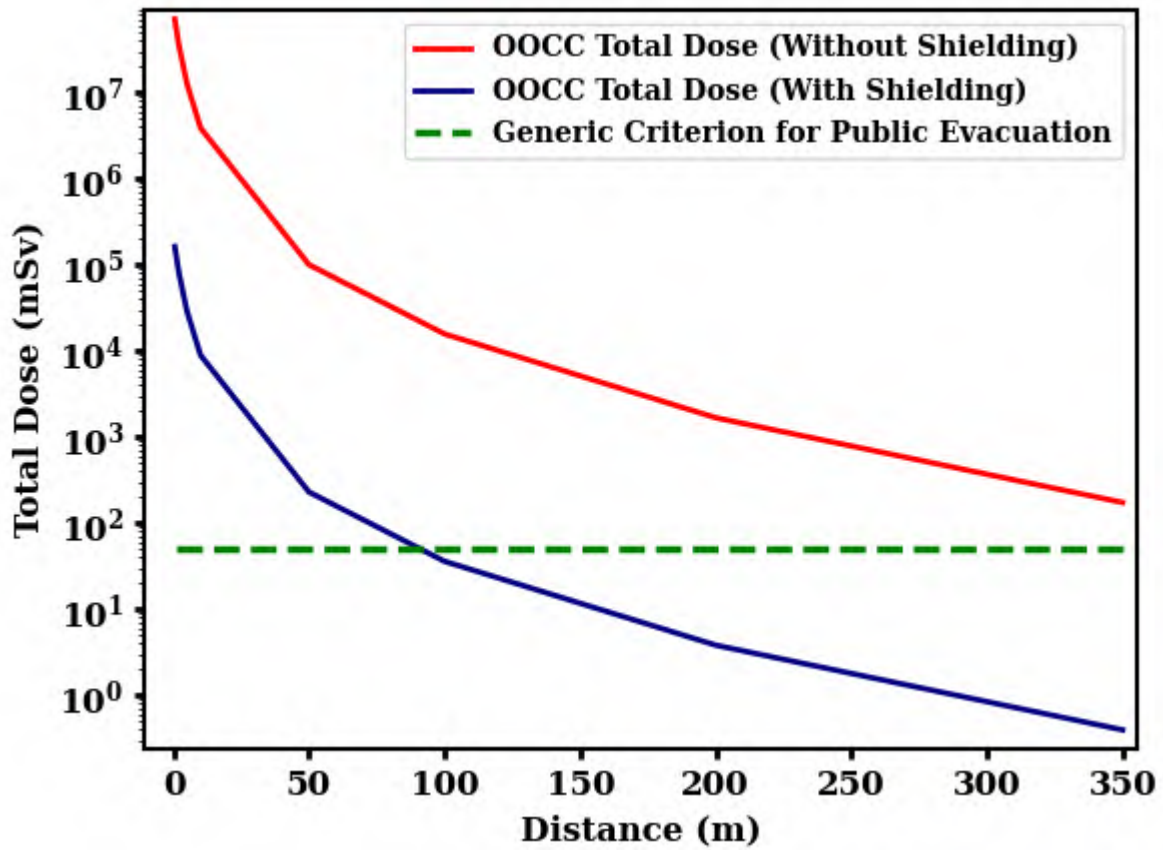
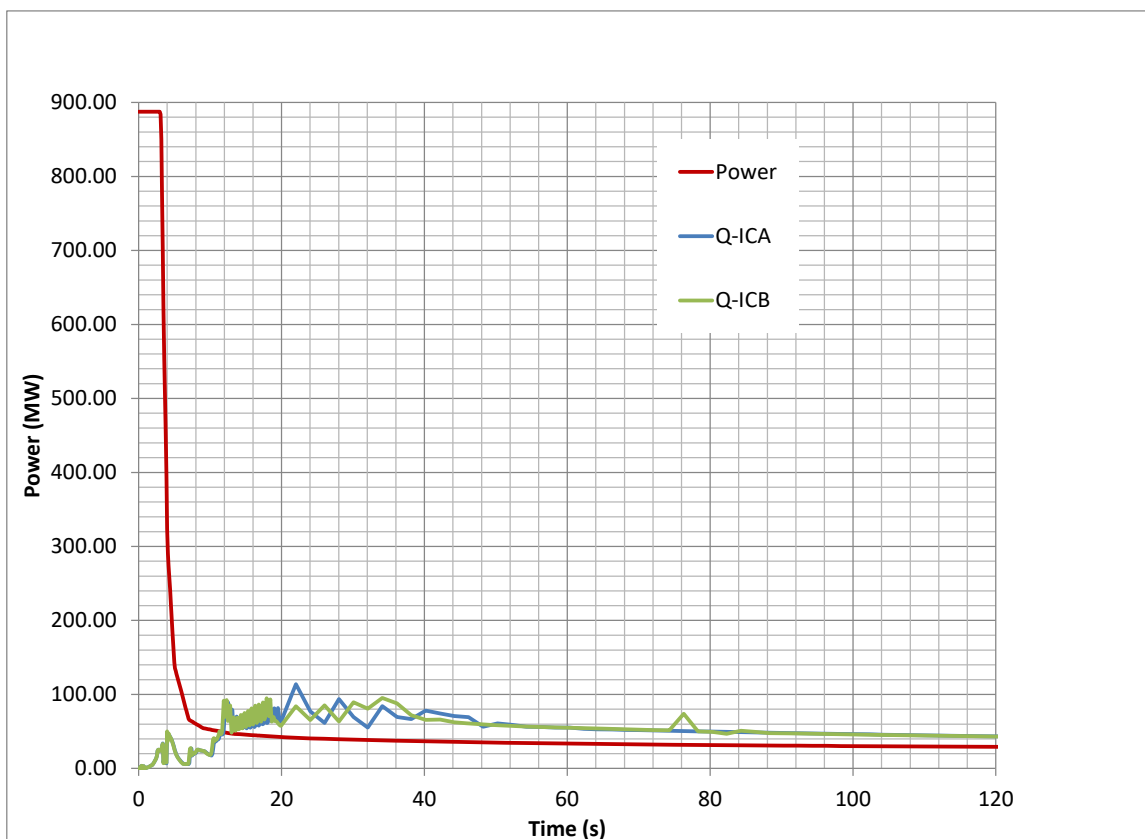
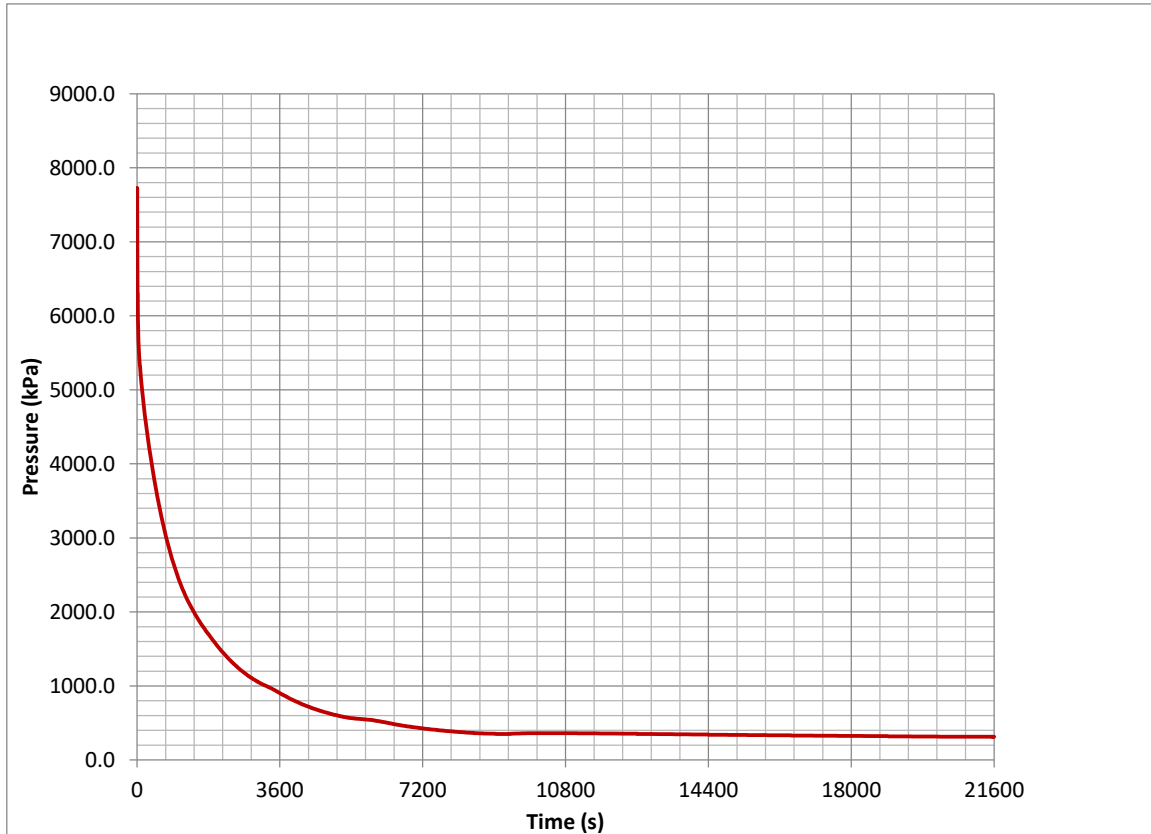


Figure 15.5-169: Plot of Out of Core Criticality Dose Consequence Versus Distance



**Figure 15.5-170: Reactor Power, Large FW Pipe Break, Conservative Case**



**Figure 15.5-171: Reactor Pressure, Large FW Pipe Break, Conservative Case**



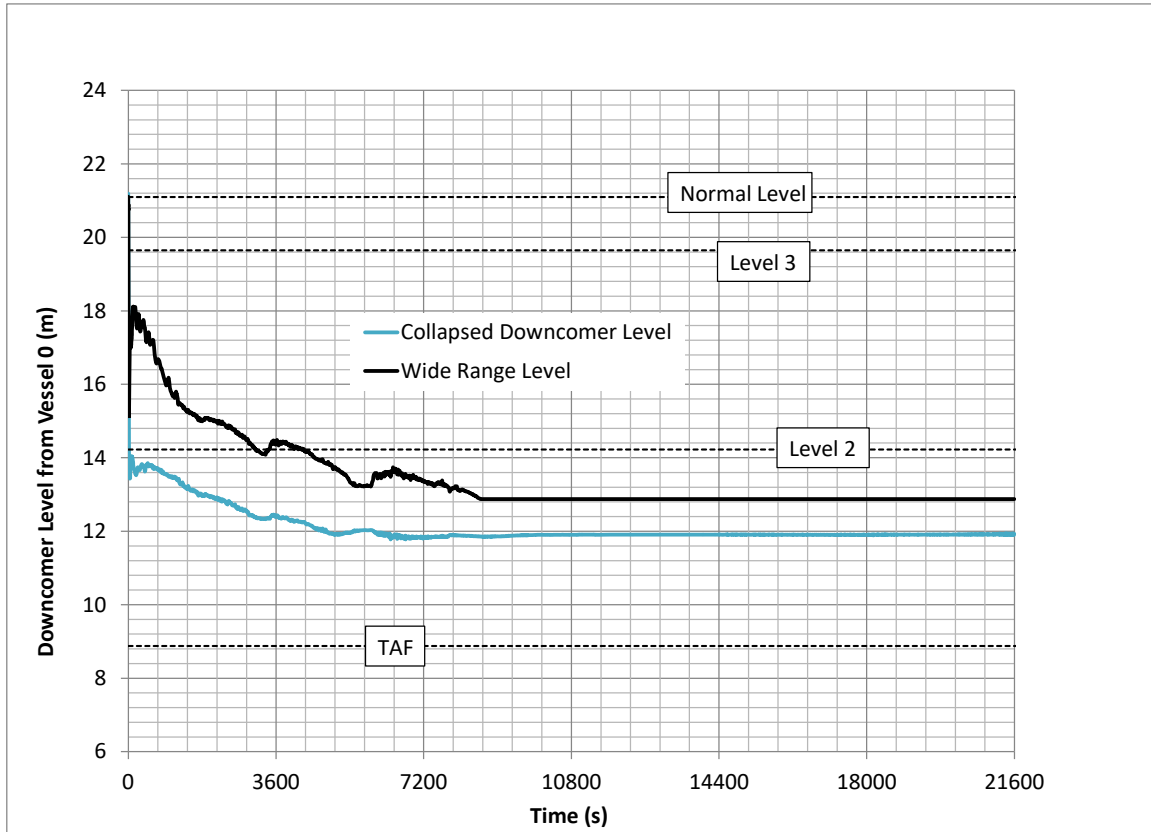


Figure 15.5-172: Reactor Water Level, Large FW Pipe Break, Conservative Case

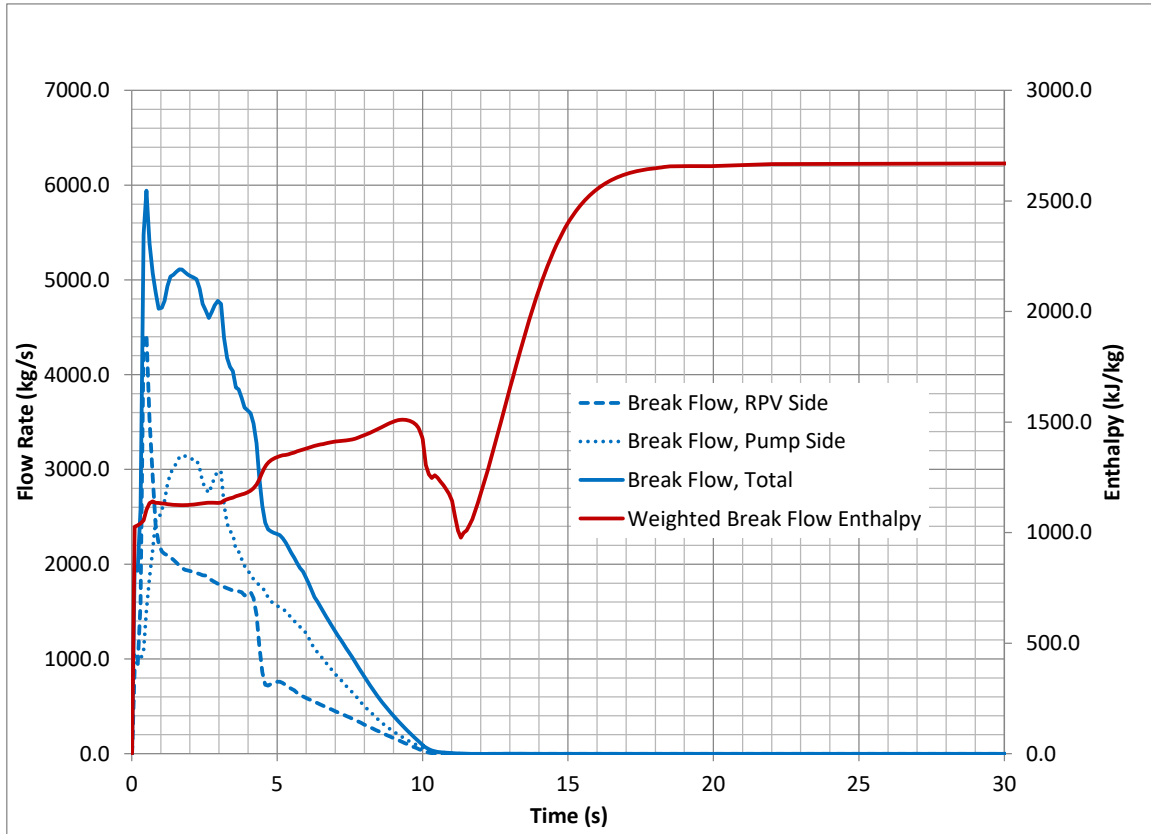
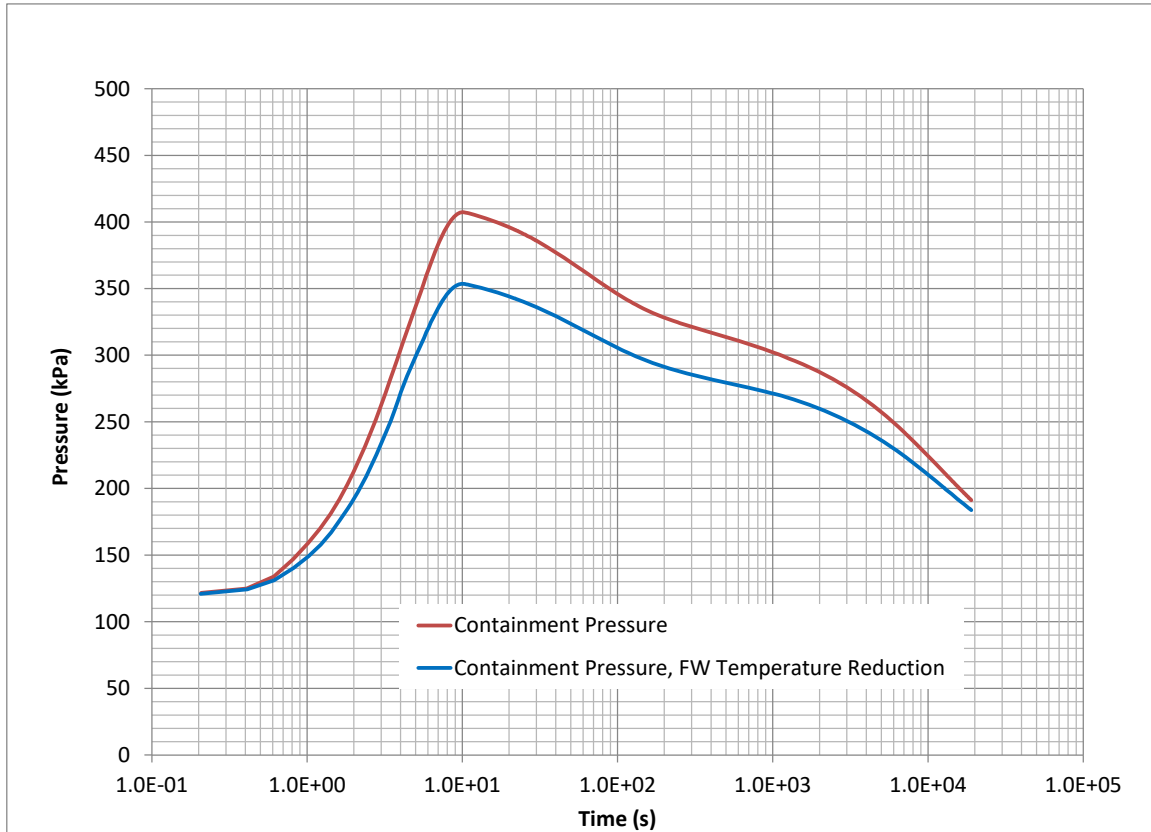
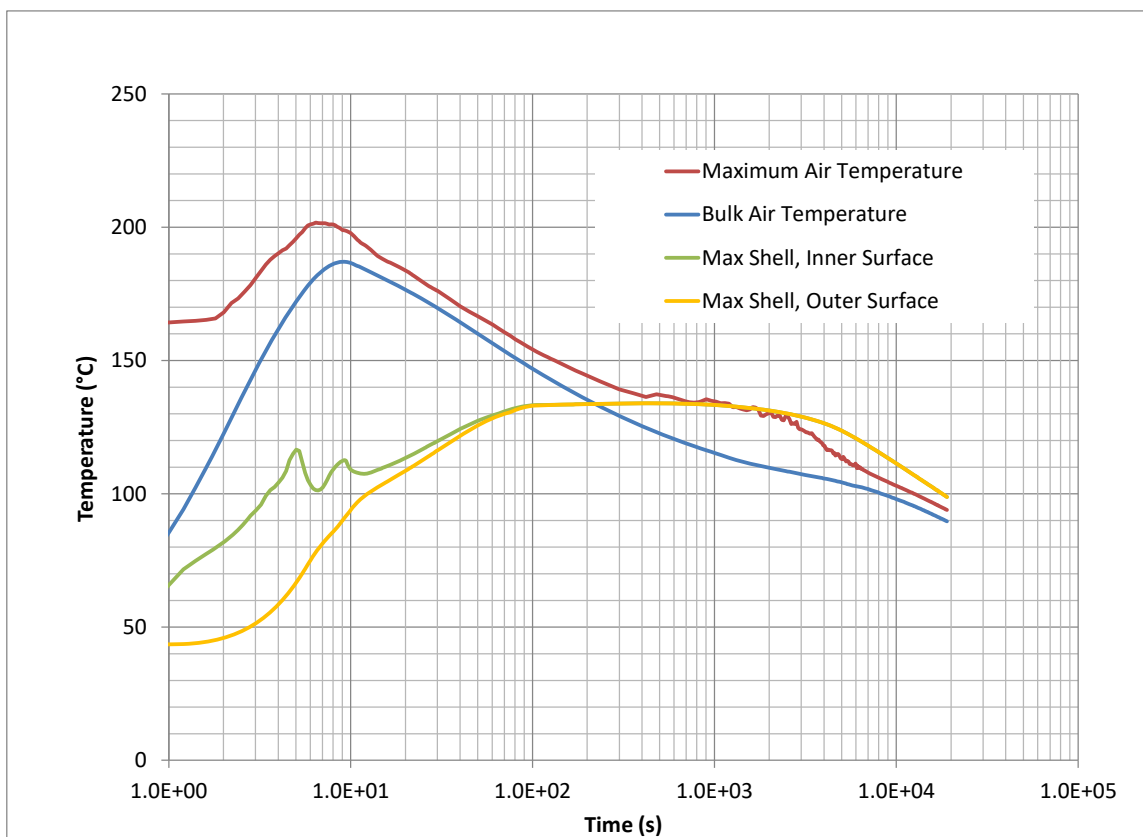


Figure 15.5-173: Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case



**Figure 15.5-174: Containment Pressure, Large FW Pipe Break, Conservative Case**



**Figure 15.5-175: Containment Temperature, Large FW Pipe Break, Conservative Case**



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33966

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 16  
Operational Limits and Conditions**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33966 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ASME	American Society of Mechanical Engineers
CNSC	Canadian Nuclear Safety Commission
D-in-D	Defence-in-Depth
LCO	Limiting Condition for Operation
NBS	Nuclear Boiler System
OLC	Operational Limits and Conditions
OPEX	Operating Experience
PSA	Probabilistic Safety Assessment
RPV	Reactor Pressure Vessel
SDC	Shutdown Cooling System
SSC	Structures, Systems, and Components



## TABLE OF CONTENTS

<b>16.0 OPERATIONAL LIMITS AND CONDITIONS .....</b>	<b>16-1</b>
16.1 Scope and Application .....	16-1
16.1.1 Use and Application .....	16-2
16.2 Bases for Development .....	16-4
16.3 Safety Limits .....	16-6
16.4 Requirements for Limits and Conditions for Normal Operation, Surveillance and Testing .....	16-7
16.5 Administrative Requirements .....	16-8
16.6 References .....	16-9
16.7 Appendix 16A - BWRX-300 Operational Modes .....	16-10
16.7.1 Modes of Normal Operation of the Plant .....	16-10

**LIST OF TABLES**

Table 16.7-1: BWRX-300 Operational Modes .....	16-12
--	-------

## **LIST OF FIGURES**

None.

## **16.0 OPERATIONAL LIMITS AND CONDITIONS**

### **16.1 Scope and Application**

The facility safe operating envelope is defined by a detailed safety evaluation (deterministic and probabilistic) and a set of limits and conditions within which the facility must be operated to ensure compliance with the safety analysis. The Operational Limits and Conditions (OLC) are those upon which reactor operation is licenced and can be monitored by or on behalf of the operator and controlled by the operator. Once the safety analysis is complete, the safe operating envelope is finalized and reflected in the Pre-Operational Safety Analysis Report in support of the Licence to Operate.

The OLC are based on the requirements of CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 16.6-1); CNSC REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility" (Reference 16.6-2); and CNSC REGDOC-1.1.3, "Licence to Operate a Nuclear Power Plant" (Reference 16.6-3). They are derived from the analyses and evaluations summarized in the safety analysis of Chapter 15. The format of the OLC is based on USNCR NUREG-1434, Volume 1, "Standard Technical Specifications – General Electric BWR/6 Plants: Specifications" and Volume 2, "Standard Technical Specifications – General Electric BWR/6 Plants: Bases" (Reference 16.6-4) to the extent practical.

The structure of the first volume (Specifications) of NUREG-1434 (Reference 16.6-4) covers Use and Application, Limiting Condition for Operation (LCO) Applicability and Surveillance Requirement Applicability, Safety Limits, Design Features, and Administrative Controls. In addition, the major portion of this first volume includes Specifications (or OLCs) in areas of Reactivity Control Systems, Power Distribution Limits, Instrumentation, Reactor Coolant System, Emergency Core Cooling Systems, Containment Systems, Plant Systems, Electrical Power Systems, Refueling Operations, and Special Operations. Each Specification OLC consists of a Limiting Condition for Operation (LCO), Applicability, Actions, and Surveillances. The Actions are included in an Action table that describes the Conditions, Required Actions, and Completion Times. The Surveillances are also included in a Surveillance table identifying the Surveillance Requirements and associated Frequency of performance. Additional details of the Specifications or OLCs are provided in Subsection 16.1.1 and Section 16.4.

Use and Application as well as LCO Applicability and Surveillance Requirement Applicability are addressed in Section 16.1.1 and provide general rules that apply to NUREG-1434 (Reference 16.6-4). Safety Limits are addressed in Section 16.3. Administrative Controls are addressed in Section 16.5. Design Features include requirements related to the site location, reactor core, and fuel storage.

The second volume includes the Bases which is further described in Section 16.2.

Each specification OLC is based on one of the following selection criteria from 10 CFR 50.36, "Technical Specifications" (Reference 16.6-5).

- Criterion 1 – Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2 – A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 3 – A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

NEDO-33966 REVISION 0  
NON-PROPRIETARY INFORMATION

- Criterion 4 – A structure, system, or component which Operating Experience (OPEX) or probabilistic risk assessment has shown to be significant to public health and safety.

Structures, Systems, and Components (SSCs) as well as initial conditions are screened against the criteria and included in the OLCs. SSCs installed to detect and indicate degradation of the reactor coolant pressure boundary are incorporated to satisfy Criterion 1. The OLCs will require the SSCs (Criterion 3) to be operable in conditions where the design basis accident or transient is postulated to occur. The OLCs will also specify requirements for initial conditions (Criterion 2) where events are postulated to occur. This will help ensure the safety criteria of the safety analyses in Chapter 15 are met if a design basis accident or transient occurs.

SSCs are evaluated against Criteria 2 and 3 using the Conservative Safety Analyses from the BWRX-300 Safety Strategy. The focus is on Safety Class 1 equipment performing the credited Defense Line 3 safety functions. The Conservative Safety Analyses addresses sequences that began with either an anticipated operational occurrence or design basis accident. The resulting consequences for these events are well within the specified safety analyses acceptance criteria.

Consistent with the Safety Strategy to include a Defence-in-Depth (D-in-D) approach, Safety Class 2 (Defense Line 4a) SSCs and Safety Class 3 (Defense Line 2) SSCs credited in the Baseline Deterministic Safety Analyses and Extended Deterministic Safety Analyses, respectively are addressed in plant programs. The program ensures Surveillances are performed to confirm proper functionality when required to mitigate transients, and to have pre-planned actions if functionality is degraded.

SSCs credited in the Probabilistic Safety Assessment (PSA) of Chapter 15 having a significant impact on Core Damage Frequency are included in OLCs to satisfy Criterion 4.

#### **16.1.1 Use and Application**

Use and application addresses definitions of important terms used in the Specifications and Bases, use of logical connectors, Completion Times for Required Actions, and Frequency of Surveillance Requirements.

Logical connectors are used in Specification or OLC to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances and Frequencies. The only logical connectors that appear are “AND” and “OR.” The physical arrangement of these connectors constitutes logical conventions with specific meanings.

Applicability LCO Applicability are provided that establish the general requirements applicable to all LCOs. LCOs Applicability can be summarized as:

- |           |   |
|-----------|---|
| LCO 3.0.1 | Compliance with an LCO is required during the operational modes or other conditions that are specified in the LCO's applicability statement.  |
| LCO 3.0.2 | Satisfying the requirements of the associated Required Actions within the specified Completion Time constitutes compliance with the Specification.  |
| LCO 3.0.3 | When an LCO and its associated Required Actions are not met, an associated Action is not provided, or if directed by the Required Actions, action must be initiated within specified time intervals to place the unit in an operational mode in which the Specification does not apply.       |
| LCO 3.0.4 | When an LCO is not met, entry into a mode or other specified condition may only be made: <ol style="list-style-type: none"><li>1. When associated actions permit continued operation in the mode or other specified condition in the applicability for an unlimited period of time,</li></ol> |

NEDO-33966 REVISION 0  
NON-PROPRIETARY INFORMATION

2. In accordance with the facility risk management program, or
3. When an allowance is stated in the individual value, parameter, or other specification.

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with actions may be returned to service under administrative control solely to perform testing required to demonstrate its operability or the operability of other equipment.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO Actions are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed to determine if a loss of safety function is determined to exist. If a loss of safety function is determined to exist, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7 Special Operations LCOs allow specified Specification requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other Specification requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the Actions of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a Mode or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

Surveillance Requirement Applicability establish the general requirements applicable to all Surveillance Requirements. Surveillance Requirement Applicability can be summarized as:

SR 3.0.1 Surveillance Requirements apply when the associated LCO applies. Upon discovery that the requirements of a Surveillance cannot be met, the LCO is not met.

SR 3.0.2 Surveillance intervals can be extended to 1.25 times the interval specified in the frequency with two exceptions:

1. If the frequency is specified as "once", the interval cannot be extended, and
2. If the performance interval applies to a Required Action that is performed on a periodic or cyclic basis, the interval cannot be extended on the first performance.

SR 3.0.3 If a Surveillance is missed and the allowed interval has expired, the affected LCO is not immediately declared not met. A delay period is allowed for the performance of the missed Surveillance. The delay may be up to 24 hours or the Surveillance interval, whichever is greater.

SR 3.0.4 An LCO's Surveillances must have been met before the unit enters a mode in which that LCO applies. When an LCO is not met due to Surveillances not having been met, entry into the Applicability is only allowed in accordance with LCO 3.0.4.

Specifications or OLCs are provided with an LCO, Applicability, Required Actions, and Surveillances that define the safe operating envelope covering the following areas:

- Reactivity Control Systems
- Power Distribution Limits

- Instrumentation
- Reactor Coolant System
- Emergency Core Cooling Systems
- Containment Systems
- Plant Systems
- Electrical Power Systems
- Refueling Operations
- Special Operations

Bases are provided for each Safety Limit and Specification that provide background information associated with the limit, discussion of the applicable safety analyses, description of the LCO, Applicability of the LCO, description of the Required Actions and associated Completion Times for each Action, and associated Surveillance Requirements that verify equipment operability or satisfaction of the LCO criteria.

The combination of information provided for use and application and the Bases for each LCO are sufficient to support training, qualification, and certification of personnel.

## **16.2 Bases for Development**

The Bases for the OLCs are provided for the various parts of Specifications. Inclusion of the Bases is required to satisfy the requirements of CNSC REGDOC-2.5.2 (Reference 16.6-1). The Bases provides information necessary for understanding and implementation of the OLC specification requirements. As such, the Bases are not part of the OLC specifications.

Because the Specifications vary in applicability and requirement, the information provided in the Bases differs accordingly providing the following:

1. Discussion of the safety limits and their relationship to the protection of fuel clad integrity, pressure boundary integrity or containment integrity.
2. A background discussion of each specification that describes the system, limit, or function as it relates to the specification including any limits that are protected by the specification and consequences of exceeding the limit.
3. An applicable safety analyses discussion that discusses the analysis and evaluation from which the specification has been derived including, as applicable, design basis accident and transient analyses, major input assumptions, relationship of the Specification to the analysis acceptance criteria, and the appropriate Specification selection criteria.
4. Discussion of the LCO or Safety Limit) with explanation of why the requirement is appropriate, why it was determined to be the lowest functional capability or performance level necessary for safe operation of the facility, the relationship to the protection of fuel clad integrity, pressure boundary or containment integrity and any other facets of the specification required such as:
  - a. Conditions required
  - b. Number of components required
  - c. Required flow paths
  - d. Parameter requirements

NEDO-33966 REVISION 0  
NON-PROPRIETARY INFORMATION

- e. Exceptions or notes
  - f. Specification violation implications
5. Discussion of the Applicability associated with the LCO with explanation of why compliance is required for the given conditions (e.g., Modes 1, 2, or 3) and why not required during other conditions (e.g., Modes 4, 5, and 6)."
  6. Discussion of Actions including:
    - a. For Required Actions to substantiate acceptability of deviation on the basis that protection is still provided, the probability of an event during the period covered or the Required Actions compensate for specification deviation
    - b. Substantiation of acceptability of Completion Times
    - c. Substantiation of Mode changes requirement and corresponding acceptability
    - d. The basis or source of all numbers in the Required Action
  7. Discussion of how the surveillances demonstrate compliance with the specification requirements including the basis of the specified acceptance criteria, justification for surveillance frequencies and parameter values, relationship to any ASME Section XI requirements (if applicable) and why the surveillance is necessary at the frequency specified.
  8. A list of references that provide more detailed information pertinent to the Specification.



### 16.3 Safety Limits

Safety Limits are limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The conditions for which the Safety Limit is applicable is provided if required. A violation of any Safety Limit requires restoration of the Safety Limit and insertion of all control rods within a specified time period.

Safety Limits are established for fuel cladding integrity, fuel cooling, and reactor coolant system pressure boundary integrity.

The fuel cladding safety limit is based on establishing a margin to the conditions that produce onset to transition boiling. Because fuel damage is not directly observable, a step back approach is used to establish the safety limit for critical power ratio. A thermal power limit is applicable when less than approximately 4.72 MPaG and a minimum critical power ratio limit when greater than approximately 4.72 MPaG.

The fuel cooling safety limit is based on establishing water level above the top of active irradiated fuel to provide core cooling with margin.

The reactor coolant system pressure boundary integrity safety limit is based on protecting the reactor coolant pressure boundary against overpressurization. In the event of fuel cladding failure and release of fission products into the reactor coolant, the reactor coolant system serves as the primary barrier in preventing fission product release into the atmosphere. Establishing an upper limit on reactor vessel steam dome pressure ensures the reactor coolant pressure boundary integrity.

## **16.4 Requirements for Limits and Conditions for Normal Operation, Surveillance and Testing**

### **Limiting Condition for Operation**

The LCO describes as simply as possible the lowest functional capability or performance level of equipment required for safe operation of the facility. Process parameters required to preserve the assumptions of the safety analysis are also the subject of LCOs. Uncertainties are considered and incorporated, such that the condition(s) specified are sufficient to provide a high degree of assurance that corresponding safety analysis limits are not exceeded.

### **Applicability**

An Applicability statement is included for each LCO that consists of a simple listing of the Modes or Conditions during which the LCO is applicable. The Modes are defined in Appendix 16A.

### **Actions**

Conditions use the term inoperable, where appropriate to describe the failure to meet an LCO. Conditions are broken down into separate statements to describe a single condition unless multiple Conditions have identical Required Actions. Conditions are ordered from the least to most degraded condition. Required Actions are established to restore the parameter to within limits, restore SSCs to operable status, or place the plant in a safe condition within specified Completion Time. The Completion Times for the Required Actions are determined based on reliability and PSA. The Completion Times are established to ensure that any increase to risk is kept to an acceptable level.

### **Surveillance Requirements**

Surveillance Requirements consist of short descriptions of each Surveillance and its Frequency of performance and are tabulated from the shortest to the longest frequency interval. Surveillance Requirements fully identify the requirements appropriate to ensure compliance with the LCO. The frequency of surveillance is based on a reliability analysis, PSA, and previous OPEX. If applicable PSA data is not available for equipment, the reliability assessment is based on supplier recommendations.

## **16.5 Administrative Requirements**

Administrative controls address provisions related to responsibility, organization, unit staff qualifications, procedures, programs and manuals, and reporting requirements, as necessary to assure operation of the facility in a safe manner.

**16.6 References**

- 16.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 16.6-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 16.6-3 CNSC Regulatory Document REGDOC-1.1.3, "Licence to Operate a Nuclear Power Plant."
- 16.6-4 USNRC NUREG-1434, Volume 1, "Standard Technical Specifications – General Electric BWR/6 Plants: Specifications," Volume 2, "Standard Technical Specifications – General Electric BWR/6 Plants: Bases."
- 16.6-5 10 CFR 50.36, "Technical Specifications."

## **16.7 Appendix 16A - BWRX-300 Operational Modes**

### **16.7.1 Modes of Normal Operation of the Plant**

The normal BWRX-300 operating modes are defined corresponding to any one inclusive combination of Reactor Mode Switch position, average reactor coolant temperature, and reactor vessel head closure stud tensioning specified in Table 16.7-1 with fuel in the reactor vessel.

#### **16.7.1.1 Mode 1: Power Operation**

Mode 1, Power Operation, is defined with the Reactor Mode Switch in the "RUN" position at any average reactor coolant temperature. The Reactor Mode Switch is not placed in "RUN" position unless the Nuclear Boiler System (NBS) is pressurized and able to turn the main turbine generator. The transition to Mode 1 normally occurs at 10%-12% rated thermal power. A plant scram occurs at approximately 15% RTP when the Reactor Mode Switch is in the Startup position.

During Power Operation, the reactor water level is controlled by the Reactor Level Control function by adjusting the speed of the operating Reactor Feedwater Pump to maintain a normal reactor water level. The reactor pressure is controlled by the Reactor Pressure Control function using reactor steam dome pressure as the input. Reactor power is controlled by the Rod Control and Information System and the Plant Automation System. Normal power maneuvering occurs by moving control rods to achieve the desired power level.

#### **16.7.1.2 Mode 2: Startup**

Mode 2, Startup, is defined with the Reactor Mode Switch in the "STARTUP" position at any average reactor coolant temperature. Startup Operation is also defined with the Reactor Mode Switch in the "REFUEL" position when all the reactor head closure studs are fully tensioned. With the Reactor Mode Switch in the "STARTUP" position additional protection is in place for low power operation however but this mode allows the reactor to start heating up and pressurizing by withdrawing control rods via the Rod Control and Information System. In the "REFUEL" position, a control rod block prevents the withdrawal of more than two control rods at a time.

#### **16.7.1.3 Mode 3: Hot Shutdown**

Mode 3, Hot Shutdown, is defined with the Reactor Mode Switch in the "SHUTDOWN" position at an average reactor coolant  $>215.6^{\circ}\text{C}$  and all Reactor Pressure Vessel (RPV) head closure studs fully tensioned. When the Reactor Mode Switch is placed into the SHUTDOWN position, a reactor scram signal is sent to the control rods. The scram signal is released after all operable control rods have been fully inserted by the hydraulic scram. Once released, the operator can reset the scram signal (if no other scram signals are present) so that the control rod drive system can recharge the scram accumulators and return flow to the reactor to the normal flow rate. A control rod withdrawal block is maintained any time the Reactor Mode Switch is in the "SHUTDOWN" position. The saturation steam temperature associated with the reactor steam dome pressure is used to determine average reactor coolant temperature if the Shutdown Cooling System (SDC) is not in service. If the SDC is in service, the higher of saturation temperature and SDC supply line temperature is used.

All reactor head closure studs are fully tensioned in this mode. This mode is entered after a reactor scram from any power level once the Reactor Mode Switch position is changed from RUN to SHUTDOWN. This mode is not typically entered during a normal transition from Refueling or Cold Shutdown to Power Operation.

#### **16.7.1.4 Mode 4: Stable Shutdown**

Mode 4, Stable Shutdown, is defined with the Reactor Mode Switch in the "SHUTDOWN" position at an average reactor coolant temperature between  $215.6^{\circ}\text{C}$  and  $93.3^{\circ}\text{C}$  and all RPV head closure

studs fully tensioned. Stable Shutdown Operation can be achieved using passive cooling systems through natural circulation. The saturation steam temperature associated with the reactor steam dome pressure is used to determine average reactor coolant temperature if the SDC is not in service. If the SDC is in service, the higher of saturation temperature and SDC supply line temperature is used. This mode is entered during a normal shutdown to Cold Shutdown. The transitions to Cold Shutdown will require active cooling systems through forced circulation. The reactor head closure studs are fully tensioned in this mode. This mode can also be entered when transitioning from Cold Shutdown to Power Operation during heat up before the hand switch has been repositioned from "SHUTDOWN" to the "STARTUP" position.

#### **16.7.1.5 Mode 5: Cold Shutdown**

Mode 5, Cold Shutdown, is defined with the Reactor Mode Switch in the "SHUTDOWN" position at an average reactor coolant temperature  $\leq 93.3^{\circ}\text{C}$  and all RPV head closure studs fully tensioned. In this mode, the temperature of the reactor coolant system is low, and the isolation condensers are not functional. Decay heat removal is accomplished through the SDC.

#### **16.7.1.6 Mode 6: Refueling**

Mode 6, Refueling, is defined with the Reactor Mode Switch in "SHUTDOWN" or "REFUEL" position, and one or more RPV head closure studs are less than fully tensioned. The reactor is in a subcritical state. In this mode the RPV cannot be pressurized, and the isolation condensers are not able to remove heat. Decay heat can be effectively transferred from the fuel and absorbed by the volume of water present in the RPV and Fuel Pool. Temperature control to allow for effective maintenance is accomplished by use of the SDC and the Fuel Pool Cooling and Cleanup System. The containment is not required to be operable during Refueling.

**Table 16.7-1: BWRX-300 Operational Modes**

<b>Mode</b>	<b>Title</b>	<b>Reactor Mode Switch Position</b>	<b>Average Reactor Coolant Temperature (°C)</b>
1	Power Operation	Run	N/A
2	Startup	Refuel <sup>(1)</sup> or Startup	N/A
3	Hot Shutdown <sup>(1)</sup>	Shutdown	> 215.6
4	Stable Shutdown <sup>(1)</sup>	Shutdown	≤ 215.6 and > 93.3
5	Cold Shutdown <sup>(1)</sup>	Shutdown	≤ 93.3
6	Refueling <sup>(2)</sup>	Shutdown or Refuel	N/A

(1) All RPV head closure studs fully tensioned.

(2) One or more RPV head closure studs less than fully tensioned.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33967

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 17  
Management for Safety**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
DNNP	Darlington New Nuclear Project
ESBWR	Economic Simplified Boiling Water Reactor
GEH	GE Hitachi Nuclear Energy
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronic Engineers
ISO	International Organization for Standardization
NIRMA	Nuclear Information and Records Management Association
OPG	Ontario Power Generation
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
QAPD	Quality Assurance Program Description
RM	Requirements Management
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components
USNRC	U. S. Nuclear Regulatory Commission

## TABLE OF CONTENTS

17.0	MANAGEMENT FOR SAFETY .....	17-1
17.1	General Characteristics of the Management System .....	17-1
17.2	Specific Elements of the Management System .....	17-4
17.2.1	Facility Development Lifecycle Phases and Activities .....	17-9
17.2.2	Facility Operational Lifecycle Phases and Activities .....	17-19
17.2.3	Facility End of Life Lifecycle Phases and Activities .....	17-20
17.3	Quality Management .....	17-21
17.3.1	Facility Development Lifecycle Phases and Activities .....	17-22
17.3.2	Facility Operational Lifecycle Phases and Activities .....	17-28
17.3.3	Facility End of Life Lifecycle Phases and Activities .....	17-28
17.4	Measurement, Assessment, and Improvement of the Owner Management System .....	17-29
17.4.1	Facility Development Lifecycle Phases and Activities .....	17-29
17.4.2	Facility Operational Lifecycle Phases and Activities .....	17-30
17.4.3	Facility End of Life Lifecycle Phases and Activities .....	17-30
17.5	Fostering a Culture for Safety .....	17-31
17.5.1	Facility Development Lifecycle Phases and Activities .....	17-32
17.5.2	Facility Operational Lifecycle Phases and Activities .....	17-33
17.5.3	Facility End of Life Lifecycle Phases and Activities .....	17-33
17.6	References .....	17-34

**LIST OF TABLES**

Table 17.2-1: OPG Nuclear Management System Programs for DNNP Development Lifecycle Activities (Site Preparation and Construction Phase) .....	17-5
---	------

**LIST OF FIGURES**

Figure 17.2-1: Contract Partner Team.....	17-10
Figure 17.2-2: Current Organizational Structure for DNNP Small Modular Reactor Organization .....	17-12
Figure 17.2-3: GEH Organizational Structure .....	17-16
Figure 17.3-1: Configuration Management Model .....	17-26

## **17.0 MANAGEMENT FOR SAFETY**

The OPG Nuclear Management System is an Integrated Management System that ensures adequate programs and processes are implemented to enable OPG to meet its objectives. OPG has experience and a proven management system framework that complies with the CSA Group (CSA) CSA N286, "Management System Requirements for Nuclear Facilities" (Reference 17.6-1) and Canadian Nuclear Safety Commission (CNSC), REGDOC-2.1.1, "Management System" (Reference 17.6-2), for its existing nuclear facilities. The OPG Nuclear Management System is established by, N-CHAR-AS-0002, "Nuclear Management System" (Reference 17.6-3). This nuclear charter gives authority to nuclear programmatic governance and describes the governing document framework. OPG will utilize applicable programs from this framework, further described throughout this chapter, that comply with associated CSA N286 (Reference 17.6-1) clauses for each nuclear lifecycle activity for the Darlington New Nuclear Project (DNNP).

### **17.1 General Characteristics of the Management System**

OPG has a mature and effective Nuclear Management System. The goal of Nuclear Management System is to establish the governance framework and processes to support OPG nuclear facilities in achieving their health, safety, environment, security, economic and quality objectives. This management system is part of OPG governance that has a hierarchical structure taking authority from OPG-POL-0032, "Safe Operations Policy" (Reference 17.6-4) and the N-POL-0001, "Nuclear Safety & Security Policy" (Reference 17.6-5).

OPG-POL-0032 (Reference 17.6-4) defines the requirement to establish and maintain management systems to ensure that safe operation is the overriding priority in all activities performed at OPG facilities. This policy outlines requirements and accountabilities for the organization to foster high levels of operating performance and reliability, while ensuring compliance with all legal and regulatory requirements. This policy then gives authority to N-POL-0001 (Reference 17.6-5).

N-POL-0001 (Reference 17.6-5) establishes guiding principles for every OPG nuclear employee, person, or entity that supplies a product or service to OPG's nuclear facilities, from initial conception of a nuclear product, through each lifecycle of the facility. This policy states that nuclear safety and security shall be the overriding priorities in all activities performed in support of OPG nuclear facilities and nuclear safety and security shall have clear priority over schedule, cost, and production. The policy requires that everyone demonstrate respect for nuclear safety and security and conduct themselves in a manner consistent with defined traits of a healthy nuclear safety and security culture. In accordance with the policy, the Chief Nuclear Officer is accountable to the Chief Executive Officer and the Board of Directors to establish a management system that establishes nuclear safety and security as the overriding priorities.

The Nuclear Management System charter, N-CHAR-AS-0002 (Reference 17.6-3), takes authority from the N-POL-0001 (Reference 17.6-5). This charter establishes the Nuclear Management System and provides direction regarding administration of nuclear licensing activities and establishes requirements to which OPG nuclear and interfacing organizations shall comply. The accountabilities of the Nuclear Management System are defined in the charter from overall accountability at the Chief Nuclear Officer level down to employees. The Chief Nuclear Officer is accountable for the establishment, implementation of, and effectiveness of the Nuclear Management System and ensures that a foundation of leadership exists to hold those within nuclear and interfacing organizations accountable for the implementation of, and adherence with, the Nuclear Management System. The charter states that every employee that executes and supports licensed activities associated with OPG nuclear facilities is responsible and shall be held accountable for complying with the expectations of the charter and referenced programs, and for

NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

ensuring their actions are deliberate and consistent with protecting, the health and safety of the workers, the public, and the environment. This responsibility also applies to every person or entity that supplies a product or service to OPG's nuclear facilities. With the support of employees and the effective implementation of the Nuclear Management System, OPG assures compliance with N-POL-0001 (Reference 17.6-5) by all who may have an impact on nuclear safety.

The charter defines the governing document framework of the Nuclear Management System with both nuclear-specific and interfacing organization programs. Program level documents that take authority from a policy or a charter describe the licensing and regulatory requirements and articulate how requirements are fulfilled. Each program is required to identify both implementing and interfacing documents. Implementing documents such as procedures or standards, that take authority from a policy or program, define processes, activities, or behavioral expectations, that can be either facility specific or apply to the entire fleet. Implementing documents also point to applicable governance support documents such as but not limited to:

- Forms
- Lists
- Instructions
- Manuals

Governance support documents that take authority from a procedure or a standard provide detailed instructions, criteria, or requirements for tasks or activities mandated by a governance document.

The OPG Nuclear Management System was established to ensure effective implementation and compliance with CNSC REGDOC-2.1.1 (Reference 17.6-2), and the requirements set out in CSA N286 (Reference 17.6-1). Both the generic and specific CSA N286-12 (Reference 17.6-1) requirements are fulfilled through the implementation of the programs within N-CHAR-AS-0002 (Reference 17.6-3) and apply to all work for OPG nuclear facilities and interfacing/supporting organizations through a graded approach commensurate with risk. Using a graded risk-based approach, all requirements apply to varying degrees depending upon the safety significance and complexity of the work being performed. If a graded approach is used, the criteria and process used for grading is defined within program governance. While safety is the most significant consideration in risk grading, other criteria may also be used, such as but not limited to consequence of error, design complexity, economics, past performance, or whether the work is First of A Kind or First In A While activity.

The DNNP is a large-scale nuclear project that is governed by the OPG Nuclear Management System. The current, PRSL 18.00/2031, "Power Reactor Site Preparation Licence" (Reference 17.6-6) and associated LCH-DNNP-PRSL 18.00 2031 "Licence Condition Handbook (LCH)" (Reference 17.6-7), require the DNNP to follow applicable programmatic elements of the OPG Nuclear Management System to comply with relevant clauses of CSA N286 (Reference 17.6-1) that support site preparation activities. The scope of site preparation activities that commences the development lifecycle activities includes:

- Construction of site access control measures
- Clearing and grubbing of vegetation
- Excavation and grading of the site
- Installation of services and utilities (domestic water, fire water, sewage, electrical, communications, natural gas) to service the future nuclear facility



NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

- Construction of administrative and support buildings inside the future protected area
- Construction of environmental monitoring and mitigation systems
- Construction of flood protection and erosion control measures

The DNNP is now preparing for the transition to the construction phase that will complete the development lifecycle of this project. The scope of the construction phase activities that will begin upon issuance of a Licence to Construct includes:

- Completion of any remaining site preparation activities
- Construction and the fuel-out commissioning of a single BWRX-300 reactor nuclear generating facility on the DNNP site
- Construction of equipment and structures, such as cooling water systems, to support up to four reactors on the DNNP site

Following completion of the construction phase, the DNNP will transition to the operational phase that will involve fuel-in commissioning, start-up, and commercial operation of the new reactor. Upon completion of the useful life of the reactor, the end of life phase will involve the decommissioning of the reactor facility. The Nuclear Management System governing the DNNP will evolve over the progression of lifecycle activities described above. Details of this progression are outlined in Section 17.2.

## **17.2 Specific Elements of the Management System**

As stated in Section 17.1, the DNNP has been incorporated into the OPG Nuclear Management System to support the requirements of site preparation activities to start the nuclear development lifecycle. The programs that DNNP is utilizing within the Nuclear Management System, as outlined in the site preparation LCH, are compliant with applicable clauses of CSA N286 (Reference 17.6-1) for site preparation activities. These programs have also been identified as those required for use by the DNNP for the Licence to Construct phase. The programs are listed in Table 17.2-1 followed by high-level descriptions of how the programs support the OPG Nuclear Management System. Further details of how the DNNP will utilize these programs are presented in Subsection 17.2.1.1.

**Table 17.2-1: OPG Nuclear Management System Programs for DNNP Development  
Lifecycle Activities  
(Site Preparation and Construction Phase)**

Document Number	Document Title
N-PROG-AS-0001	Nuclear Management System Administration (Reference 17.6-8)
N-PROG-AS-0002	Human Performance (Reference 17.6-9)
N-PROG-MP-0009	Design Management (Reference 17.6-10)
N-PROG-MP-0014	Reactor Safety Program (Reference 17.6-11)
N-PROG-RA-0001	Consolidated Nuclear Emergency Plan (Reference 17.6-12)
N-PROG-RA-0002	Conduct of Regulatory Affairs (Reference 17.6-13)
N-PROG-RA-0003	Performance Improvement (Reference 17.6-14)
N-PROG-RA-0010	Independent Assessment (Reference 17.6-15)
N-PROG-RA-0011	Nuclear Security (Reference 17.6-16)
N-PROG-RA-0015	Safeguards and Nuclear Material Accountancy (Reference 17.6-17)
N-PROG-TR-0005	Training (Reference 17.6-18)
OPG-PROG-0001	Information Management (Reference 17.6-19)
OPG-PROG-0005	Environment Health and Safety Managed Systems (Reference 17.6-20)
OPG-PROG-0009	Items and Services Management (Reference 17.6-21)
OPG-PROG-0037	OPG Business Planning (Reference 17.6-22)
OPG-PROG-0039	Project Management (Reference 17.6-23)
OPG-PROG-0042	Cyber Security (Reference 17.6-24)
W-PROG-WM-0003	Decommissioning Program (Reference 17.6-25)

### **Nuclear Management System Administration**

N-PROG-AS-0001 (Reference 17.6-8) describes the integration of programs and processes established by OPG to demonstrate effective implementation and compliance with the requirements set out in CSA N286 (Reference 17.6-1). Programs and processes are created such that all applicable regulatory requirements and codes and standards are embedded and integrated within the Nuclear Management System including aspects of health, safety, environment, security, economics, and quality. Reviews are performed on programs within the Nuclear Management System for their overall effectiveness and opportunities for improvement. This program also ensures that Program Owners and Authorization Authorities understand their role in maintaining an effective management system, and that any Nuclear Management System processes are standardized to the greatest extent possible.

### **Human Performance**

N-PROG-AS-0002 (Reference 17.6-9), establishes a systematic framework for human performance management. This program describes key accountabilities, core processes and related activities associated with the conduct of human performance management across all facets of nuclear within OPG. The goal of the program is to continually reduce the frequency and severity of events through the systematic reduction of human error and the management of defences in pursuit of zero events of consequence.

### **Design Management**

N-PROG-MP-0009 (Reference 17.6-10), provides a framework which ensures that changes to plant design are controlled in a manner consistent with plant design and licensing basis. This program provides assurance that design bases, design outputs and design process documentation are prepared, reviewed, approved, and implemented in accordance with approved procedures, applicable regulatory requirements, standards, and industry practices. This program also allows for design agencies to complete work following the Design Management procedures or to approved procedures taking authority from a quality program that has been approved by OPG.

### **Reactor Safety Program**

N-PROG-MP-0014 (Reference 17.6-11), defines the program elements and organizational responsibilities for the implementation and management of Nuclear Safety Analysis and major components of safe operation. It governs Safety Analysis Basis, Safe Operating Envelope, Beyond Design Basis Accident Management, and Nuclear Waste Management Facilities Safety Reports and Safety Analysis. The development and maintenance of program outputs allow OPG to demonstrate management of nuclear safety throughout the life of its nuclear facilities.

### **Consolidated Nuclear Emergency Plan**

N-PROG-RA-0001 (Reference 17.16.12), implements and maintains OPG's emergency response capability to protect the public, employees, and environment in the event of a nuclear emergency. It provides a framework for interaction with external authorities and defines OPG commitments under the Provincial Nuclear Emergency Response Plan. It also represents a basis for controlling changes and modifications to the OPG Nuclear Emergency Preparedness program.

### **Conduct of Regulatory Affairs**

N-PROG-RA-0002 (Reference 17.6-13), ensures OPG complies with the regulatory requirements in an effective and efficient manner. The program describes the procedures related to licensing, regulatory interpretations, event reporting, regulatory approvals, CNSC inspections, issue management, and communications. This includes broad guidelines for evaluating the impact of licence and permit applications and amendments, providing regulatory and legislative interpretations, and monitoring emerging legislative, regulatory, and industry trends. Successful interface with regulatory agencies is critical in meeting OPG's overall objective for its nuclear fleet. The program defines a set of processes to ensure these expectations are met in an effective and efficient manner.

### **Performance Improvement**

N-PROG-RA-0003 (Reference 17.6-14) consists of organizational learning tools and provides a framework to guide development, assessment, and improvement of the performance of OPG nuclear facilities. The framework of processes includes a means to prevent, detect and correct adverse conditions as well as facilitate the communication of lessons learned. This program covers the key areas of performance improvement, namely, corrective action, self-assessment, benchmarking, operating experience, and nuclear safety culture.

### **Independent Assessment**

N-PROG-RA-0010 (Reference 17.6-15), provides independent assessment processes, in order to perform comprehensive and critical evaluation of all activities affecting OPG nuclear facilities. This program also ensures the Nuclear Management System under N-CHAR-AS-0002 (Reference 17.6-3), is reviewed with sufficient frequency to confirm its continuing effectiveness.

### **Nuclear Security**

N-PROG-RA-0011 (Reference 17.6-16), supports the protection of nuclear assets at OPG in accordance with the legislative requirements, OPG-POL-0032 (Reference 17.6-4) and N-POL-0001 (Reference 17.6-5). This program supports the need to protect nuclear assets by:

- Establishing an enhanced state of security readiness, as a result, of potential terrorism and commitments to the CNSC
- Maximizing the response capability to contain, mitigate, and terminate a security event that has either occurred or is in progress
- Minimizing the adverse impact on nuclear plant staff or operations

### **Safeguards and Nuclear Material Accountancy**

N-PROG-RA-0015 (Reference 17.6-17), puts special controls in place to ensure OPG complies with Canada's commitment to the Nuclear Non-proliferation Treaty consistent with International Atomic Energy Agency (IAEA) requirements for storage and handling of nuclear fuel throughout its lifecycle. The program includes the following:

- Communication protocol between the IAEA, CNSC, and Nuclear
- Processes and reporting to meet applicable regulatory requirements and requirements of associated safeguards procedures

## **Training**

N-PROG-TR-0005 (Reference 17.6-18), applies to regular staff, contractors, temporary personnel, and other staff assigned work under OPG's Nuclear Management System. The program provides the structure, processes, and tools for defining, developing, implementing, documenting, assessing, and improving the training required to ensure nuclear staff have the appropriate knowledge, skill, and attitudes.

## **Information Management**

The Information Management Program, OPG-PROG-0001 (Reference 17.6-19), establishes a set of standards and procedures for the management of OPG's information throughout its lifecycle, regardless of media, including systems such as email, SharePoint, and the Intranet to ensure consistent and appropriate use. It establishes processes for the management, maintenance and final disposition of records and documents, as well as the process for the issuance, distribution, and maintenance of governance.

## **Environment Health and Safety Managed Systems**

OPG-PROG-0005 (Reference 17.6-20), establishes process requirements implemented and maintained in order to ensure OPG activities are conducted in a manner that prevents or mitigates conventional health and safety risks to workers, and/or adverse impacts on the natural environment. This program encompasses OPG's Environmental Management System and Health and Safety Managed System and outlines the associated roles and responsibilities within the organization.

## **Items and Services Management**

OPG-PROG-0009 (Reference 17.6-21), supports supply chain activities across OPG. This program ensures that procurement is planned, and that purchased, stored, and issued items and purchased services meet appropriate design and quality requirements. The program also ensures that items, services, and nuclear fuel are purchased in accordance with requirements and controlled through proper identification, receipt, inspection, handling, storage, issuance, and shipping to ensure quality of equipment and components is preserved.

## **OPG Business Planning**

OPG-PROG-0037 (Reference 17.6-22), establishes the framework within which OPG conducts its annual business planning process. This process directs the organization's resources to meet strategic goals aligned with the company's objectives. The program also describes the chief elements and processes as well as the roles and accountabilities of the groups involved and provides a structure for undertaking business planning activities at OPG. Business plans produced through this program reflect prioritized risks to achieve required objectives and include resources to implement applicable risk treatment plans.

## **Project Management**

OPG-PROG-0039 (Reference 17.6-23), sets out the principles and requirements for planning, organizing, and managing resources to ensure the safe, consistent, effective execution and completion of all projects within OPG. The program is structured to ensure that all projects are managed in a consistent manner in accordance with best practices for project management, interfacing corporate governance, and specific requirements of CSA N286 (Reference 17.6-1) related to project management.

## **Cyber Security**

OPG-PROG-0042 (Reference 17.6-24), establishes a framework identifying the processes and overall requirements for an effective cyber security program that supports the protection of cyber assets at OPG from cyber-attacks. This program also ensures protection of nuclear cyber essential assets.

## **Decommissioning Program**

W-PROG-WM-0003 (Reference 17.6-25), describes controls for the decommissioning of OPG's owned nuclear facilities and provides assurance that work will be planned and controlled in accordance with regulatory requirements. This program outlines the organizational responsibilities, interfaces, and key program elements, requirements, and processes for decommissioning.

### **17.2.1 Facility Development Lifecycle Phases and Activities**

#### **17.2.1.1 OPG's Management System**

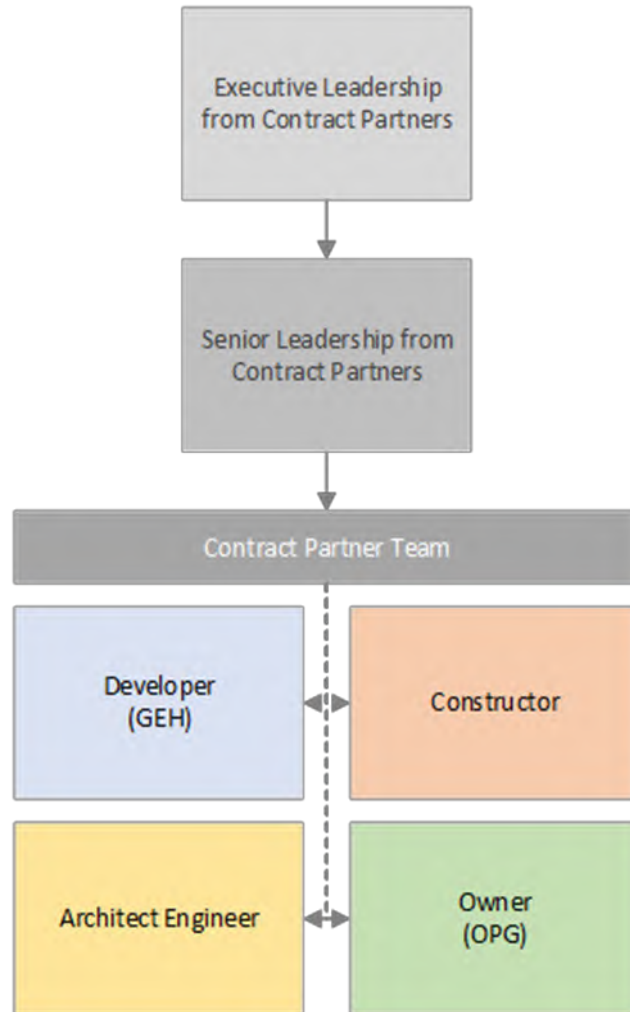
The DNNP will utilize the OPG Nuclear Management System programs listed in Section 17.2 to the extent required of a large-scale nuclear project for the development lifecycle phase. Further details of the use of these Nuclear Management System programs and the basis for selection are described in the, NK054-REP-08130-00004, "DNNP Licence to Construct Management System Report" (Reference 17.6-26).

## **DNNP Program Management**

Following the requirements of OPG-PROG-0039 (Reference 17.6-23), the DNNP has a project governance structure that is described in the NK054-PLAN-01210-00008, "Darlington New Nuclear Project – Program Management Plan" (Reference 17.6-27). This project governance structure includes a project charter, Program Management Plan and Project Management Plans. The Project Management Plan details how DNNP is managed to meet the intent of OPG's Nuclear Management System while establishing program-specific requirements. These plans will be revised routinely as the project progresses to support business planning or significant project milestones. The Project Management Plans describe aspects related to specific sub-projects within the DNNP such as contracting, resourcing, stakeholder management, etc.

## **Project Contract Model**

For the DNNP development lifecycle phase, OPG will utilize a contract model which maximizes integration and collaboration with other contract partners involved with this phase. The contract agreement will describe the relationship and accountabilities of the contract partners including owner, developer, constructor, and Architect Engineer. A visual representation of the contract partner team is provided in Figure 17.2-1 below.



**Figure 17.2-1: Contract Partner Team**



NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

The roles within the contract partner team are further described below:

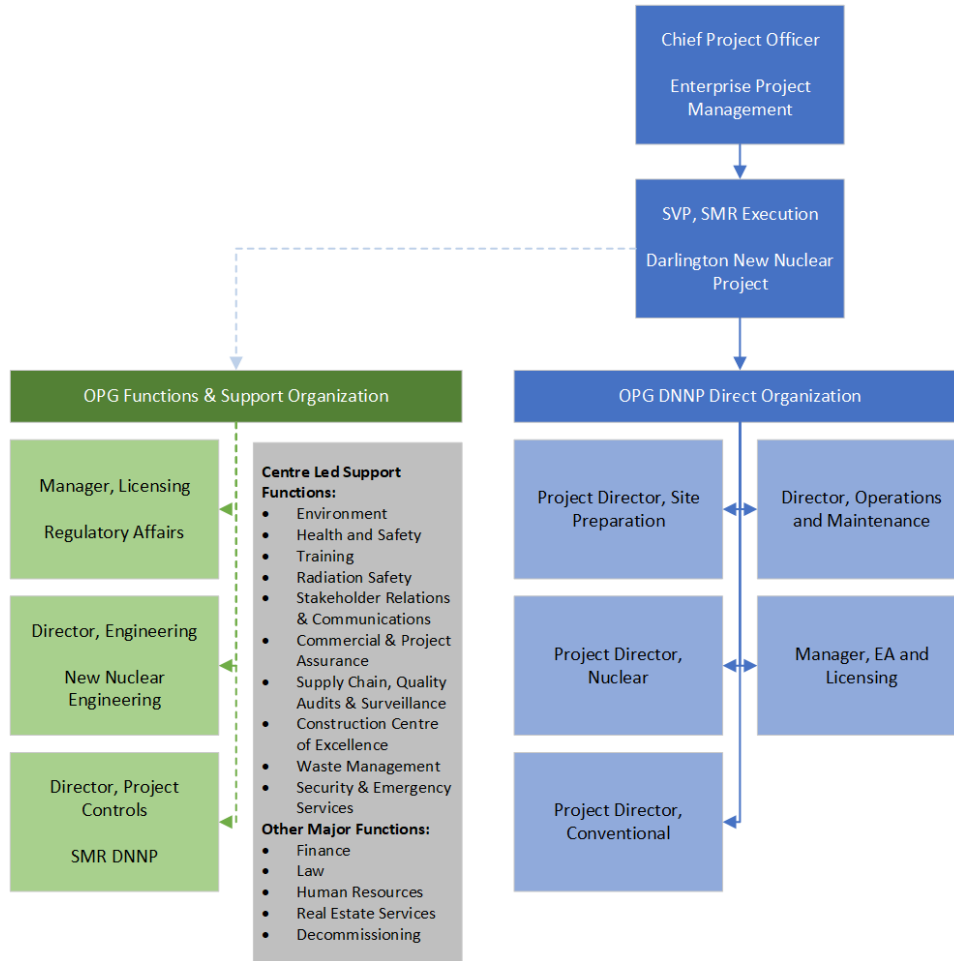
1. The developer (GEH) is the Design Authority for the power block, intake structure, Condenser Cooling Water, and switchyard scope of the project. Activities to be performed by GEH include design, procurement, and commissioning support in accordance with their managed system.
2. The Architect Engineer is responsible for development of the conventional aspects of the detailed design work. The activities associated with this scope include design, procurement, and commissioning support in accordance with their managed system and oversight by GEH as the Design Authority, and OPG as the owner.
3. The constructor is responsible for activities such as procurement, construction, and commissioning that will be performed in accordance with their managed system.
4. The owner (OPG) is also the licensee and is accountable to maintain intrusive oversight and project management of all aspects of the project with increased focus when nuclear safety could be impacted. OPG's Nuclear Management System is utilized to provide the governance, for ensuring contract partners are qualified to perform designated scope throughout the lifecycle of the project per CSA N286 (Reference 17.6-1) requirements. OPG is an informed customer that has developed and continues to increase the in-house knowledge required to provide on-going oversight and technical challenge to ensure licence and regulatory requirements are met. OPG will also retain Design Authority role for areas outside of GEH Design Authority scope, which may involve OPG taking part in similar activities listed above, such as procurement, commissioning support, as applicable. OPG will regain Design Authority for the entire plant prior to start-up.

All contract partners are required to meet the applicable requirements of CSA N286 (Reference 17.6-1) for the activities they will perform. The contract agreement provides information on the applicable management system governance and processes to be met for the licensed construction activities, with OPG continuing to use its Nuclear Management System for oversight and maintain ultimate accountability as the licensee. Project decision making will be prioritized at the appropriate level with escalation of decisions if all contract partner members are not able to agree.

### **OPG's Organization**

OPG has established a project team to manage and oversee the planning and development of the DNNP for the construction of the first Small Modular Reactor (SMR). The organizational structure for the DNNP SMR is presented in Figure 17.2-2. This organizational structure is established to optimize intrusive oversight and project management of all parties for DNNP licensed site preparation and construction activities. It aligns with the contract model which allows for increased collaboration between the owner (OPG), developer (GEH), constructor and Architect Engineer. Owner oversight is maintained separately from the contract partners to ensure independence and ultimate accountability as the licensee. The project team reports under Enterprise Projects to the Chief Project Officer at OPG and contains project management and control staff of the work scope stated in Section 17.1. The project team is supported by dedicated staff that report to fleet organizations. A focus on nuclear safety is maintained throughout the project organization with accountability as the licensee to ensure nuclear safety is maintained as the top priority for all contract partners.

NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION



**Figure 17.2-2: Current Organizational Structure for DNNP Small Modular Reactor Organization**

## **Authority Roles**

At all times, OPG will retain overall accountability for the licensed activities conducted during the development lifecycle phase. As stated previously, the Developer-GEH is the Design Authority of the power block (nuclear and conventional systems in the powerhouse), Condenser Cooling Water, the intake structure, and switchyard systems and structures during the design, construction, inspection and testing, and fuel-out commissioning phases. OPG will provide oversight in the areas of nuclear safety, safe operating envelope, design bases, and licensing basis. Interface requirements between GEH Design Authority scope and OPG Design Authority (DNNP Engineering Director) scope and accountabilities shall be determined to ensure the division of responsibilities between Design Authority scope interfaces. This will be detailed in documents such as a Contract owner Interface Requirements document and detailed interface specifications. Collaborative planning will be required to ensure appropriate input, specifications, communication, Quality Assurance (QA), oversight, and approvals are provided and documented for interfacing systems for both Design Authorities. OPG will retain Design Authority for areas outside of GEH Design Authority scope and be accountable for interface management between the power block and the remaining site works. A turnover strategy describes the transition of the GEH Design Authority to OPG Design Authority prior to plant start-up as per CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," Section 5.1 (Reference 17.6-28). Design documents will be issued into the OPG approved information management system as part of the Design Authority turnover.

The constructor will lead construction activities with support from GEH and oversight from OPG. Further details of construction and commissioning activities can be found in Chapter 14.

The Operational Authority resides with OPG and authorities of the constructor including site security and health and safety management will transition to the Operational Authority during the construction phase. Further operations aspects of the DNNP can be found in Chapter 13.

## **Oversight**

OPG DNNP Engineering must ensure appropriate interfaces and oversight protocols, as required by N-PROG-MP-0009 (Reference 17.6-10), are maintained throughout the design, procurement, installation, commissioning, and closeout phases. This is required to ensure awareness of activities and decisions in the areas of nuclear safety, safe operating envelope, design basis, and licensing basis. Additionally, this ensures compliance with design requirements, owner's requirements, and managed systems. Level of oversight, support, and depth of planning activities follow a graded approach proportional with the level of risk to DNNP, the complexity of the design, the design's effect on safety, and the past performance of the contractor(s) involved.

Additional intrusive oversight will be provided on interfacing areas between GEH Design Authority and OPG Design Authority. Oversight activities may include but are not limited to:

- Frequent design meetings with GEH at working level
- Review and input on design documentation by Subject Matter Experts
- Completion of oversight reports on Structures, Systems, and Components (SSC)
- Challenge meetings at the end of design phases
- Documented acceptance processes

Non-engineering governance and oversight is governed and performed by other executing organizations, as defined in NK054-PLAN-01210-00008 (Reference 17.6-27) and various Project Management Plans.

As GEH is currently under contract with OPG to perform development lifecycle activities for the DNPP, a description of the GEH management system is provided in Subsection 17.2.1.2.

#### **17.2.1.2 GEH Management System**

GEH is committed to the safety of its employees and the public above all other demands. GEH is committed to achieving the highest quality products and services through the disciplined application of the GEH Quality Management System.

GEH maintains an Integrated Management System documented in the Quality Assurance Program Description (QAPD) and its associated implementing procedures. Together they provide for control of GEH activities that affect the quality of nuclear plant SSC and include all planned and systematic activities necessary to provide adequate confidence that such SSC will perform satisfactorily in service.

The QAPD is divided into the following parts:

- Part I: Introduction
- Part II: QAPD Details
- Part III: Supplemental Details
- Part IV: Regulatory Commitments

**Part I** provides a high-level description of the scope and applicability of the QAPD.

**Part II** describes the portion of the Quality Management System that is required to meet the regulatory requirements. Specifically, it applies to activities involving nuclear facility (e.g., nuclear power plants and fuel reprocessing plants) SSC that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Part II has been developed to comply with regulatory requirements of the United States of America, specifically 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (Reference 17.6-29) 10 CFR Part 71 Subpart H, "Quality Assurance," (Reference 17.6-30) 10 CFR Part 21, "Reporting of Defects and Noncompliance," (Reference 17.6-31) and American Society of Mechanical Engineers (ASME) NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 17.6-32). Part II is structured in 18 sections that align with 10 CFR 50, Appendix B (Reference 17.6-29) criteria and ASME NQA-1 (Reference 17.6-32) requirements (Reference 17.6-32).

Each of the 18 sections are described below.

**Part III** describes the supplemental portions of the Quality Management System that are not specifically addressed by 10 CFR 50, Appendix B (Reference 17.6-29) or ASME NQA-1 (Reference 17.6-32) to create an integrated quality program for projects that may refer to ISO 9001, "Quality Management Systems - Requirements" (Reference 17.6-33), IAEA GSR Part 2, "Leadership and Management for Safety" (Reference 17.6-34), KTA-1401, "General Requirements for the Quality Assurance" (Reference 17.6-35), or other standards as applicable.

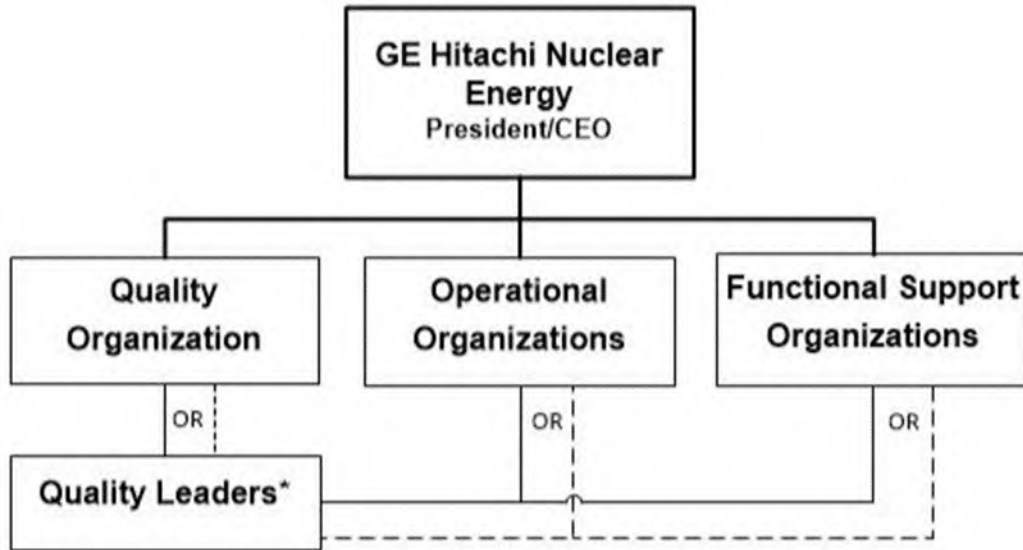
NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

**Part IV** lists the regulations, standards, and guidance that General Electric companies are committed to comply with through the implementation of the QAPD and its associated implementing procedures.

The GEH management system has been assessed for compliance with the applicable elements of CSA N286 (Reference 17.6-1), CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 17.6-36), and CSA N299.1, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 1," (Reference 17.6-37). Mapping is documented in Gap Analysis of GEH Quality Management System for Compliance to CSA N286 (Reference 17.6-1), CSA N286.7 (Reference 17.6-36) and CSA N299.1 (Reference 17.6-37) and are addressed through the implementation of GEH Project Quality Plan for DNNP described in Section 17.3.1.2.

#### Section 1 – Organization

The QAPD, Section 1, establishes the requirements for the organizational structure, functional responsibilities, levels of authority and interfaces for establishing, executing, and verifying the QAPD. The GEH organizational structure for the purposes of managing its Quality Management System is shown in Figure 17.2-3.



\*May report directly or indirectly to SVP of Quality for matters affecting Nuclear Quality; Ref. QAPD Section 1.4.2.6

**Figure 17.2-3: GEH Organizational Structure**

**Operational Organizations** are segments of the broader GEH organization that are established to comply with the regulatory requirements and are responsible for:

- Project management
- Product and services management and delivery
- Continued plant operations support
- Development of new plants, technologies, and related services

Functional Support Organizations are responsible for facilitating and supporting the activities of the Operational Organizations. Functional Support Organizations perform key functions such as quality, Engineering, Manufacturing Operations and Sourcing. Additional support functions include, but are not limited to, Sales, Commercial Operations, Finance, Human Resources, Legal/Compliance, and Government Affairs.

The Quality Organization is responsible for managing and supporting implementation of the Quality Management System and the QAPD with responsibility for quality programs and quality oversight.

Quality Leaders are responsible for supporting the implementation of the quality requirements set forth in this QAPD by verifying that administrative controls are implemented or by assuring that an activity has been correctly performed within their assigned organizations. Verification and assurance activities may include the supervision and oversight of quality engineers, inspection personnel, and the inspections or verification activities performed by independent parties. Quality Leaders typically report to the Quality Organization and indirectly to the Operational or Functional Organizations but, if required, may report directly to the Operational or Functional Organizations.

## Section 2 – Quality Assurance Program

The QAPD, Section 2, establishes the requirements for the QA program, as documented through the QAPD and GEH governance framework.

The GEH governance framework is a hierarchal set of documents known as ‘GEH Procedure Hierarchy’ that is comprised of Defining Documents, Implementing Documents, and Supporting Documents described as follows:

1. **Defining Documents** are the highest tier of documents that establish requirements and commitments. Examples include policies, directives, process descriptions, manuals, and plans.
2. **Implementing Documents** are the middle tier of documents that include steps or instructions that must be executed to meet requirements or commitments. Examples include procedures and work instructions.
3. **Supporting Documents** are the lowest tier of documents that supplement Defining or Implementing Documents that provide and offer content enhancement or clarifying instruction. Examples include forms, charters, guides, handbooks, job aids, lists, templates, and training materials.

### Section 3 – Design Control

The QAPD, Section 3, establishes the requirements for design control including the control of design inputs, design processes, design analyses, design verification, change control, interface control, and documentation and records.

The goal of design control is to ensure the design of nuclear facilities and their associated design documents meet all applicable technical requirements, regulatory requirements, codes and standards, and contractual requirements.

### Section 4 – Procurement Document Control

The QAPD, Section 4, establishes the requirements for ensuring that sufficient technical, quality, and regulatory requirements are included or referenced in the documents for procurement of items and services.

### Section 5 – Instructions, Procedures, and Drawings

The QAPD, Section 5, establishes the requirements to ensure that activities affecting quality are prescribed by and performed in accordance with implementing procedures or work instructions.

### Section 6 – Document Control

The QAPD, Section 6, establishes the requirements to ensure the control of the preparation, issuance, and changes to documents that specify quality requirements or prescribe how activities affecting quality, including organizational interfaces, are controlled to ensure that correct documents are being employed.

### Section 7 – Control of Purchased Material, Equipment and Services

The QAPD, Section 7, establishes the requirements to ensure that purchased items and services meet specified requirements.

### Section 8 – Identification and Control of Materials, Parts, and Components

The QAPD, Section 8, establishes the requirements to provide for the identification and control of materials, parts, and components.

### Section 9 – Control of Special Processes

The QAPD, Section 9, establishes the requirements for the control of special processes such as welding, heat treating, and nondestructive examination.

### Section 10 – Inspection

The QAPD, Section 10, establishes the requirements for the inspection of items and activities to verify conformance to requirements and adherence to documented instructions, procedures, and drawings.

### Section 11 – Test Control

The QAPD, Section 11, establishes the requirements to control testing, to collect data such as for design input or to verify conformance of an item to specified requirements, or to demonstrate satisfactory performance for service.

### Section 12 – Control of Measuring and Test Equipment

The QAPD, Section 12, establishes the requirements, including the procedures, to control the calibration, maintenance, storage, and use of Measuring and Test Equipment.



#### Section 13 – Handling, Storage, and Shipping

The QAPD, Section 13, establishes the requirements to control the handling, storage, packaging, cleaning, shipping, and preservation of items to prevent inadvertent damage or loss, and to minimize deterioration.

#### Section 14 – Inspection, Test and Operating Status

The QAPD, Section 14, establishes the requirements to identify the inspection and test status of individual items and for indicating the operating status of equipment, such as valves and switches. Implementing procedures specify the authority for the application and removal of these status indicators.

#### Section 15 – Control of Nonconforming Items

The QAPD, Section 5 establishes the requirements to control nonconforming items and to prevent inadvertent installation or use.

#### Section 16 – Corrective Action

The QAPD, Section 16, establishes the requirements to establish the necessary measures to promptly identify, control, document, classify, and correct Conditions Adverse to Quality.

#### Section 17 – Quality Assurance Records

The QAPD, Section 17, establishes the requirements to provide provisions for the identification, administration, receipt, storage, preservation, safekeeping, retrieval, and disposition of all records such that damage or loss does not occur.

#### Section 18 – Audits

The QAPD, Section 18, establishes the requirements to verify compliance to QA program requirements, to verify that performance criteria are met, to determine the effectiveness of the program, and to provide a comprehensive independent evaluation of activities and procedures through an established audit program.

### **17.2.1.3 Constructor and Architect Engineering Firm Management System**

Additional organizations outside of OPG or GEH contracted to perform work for the DNNP, such as the Constructor and Architect Engineering Firm, will be required to meet applicable clauses of CSA N286 (Reference 17.6-1) based on their scope of work as defined in associated contractual agreements.

Oversight of such requirements by OPG, will be performed through the processes outlined in OPG-PROG-0009 (Reference 17.6-21).

### **17.2.2 Facility Operational Lifecycle Phases and Activities**

The operational lifecycle phase for the DNNP will consist of activities such as the loading of nuclear fuel, fuel-in commissioning, start-up, and commercial operation of the nuclear facility. During this lifecycle phase, operational programmatic governance will be required to be in place prior to use. Analysis will be completed and included as part of the Licence to Operate application to ensure this operational governance will meet all applicable regulatory requirements, as well as meet the needs of the BWRX-300 reactor technology and facility organizational structure. Operational programmatic governance will include the creation of an Out of Core Criticality program that is a requirement from CNSC REGDOC-2.4.3, “Nuclear Criticality Safety” (Reference 17.6-38).

### **17.2.3 Facility End of Life Lifecycle Phases and Activities**

The end of life lifecycle phase for the DNNP will consist of decommissioning the nuclear facility. Analysis will be completed and included as part of the Licence to Decommission application to ensure that required programmatic governance will meet all applicable regulatory requirements for this lifecycle phase. If vendors are involved during this phase, similar to the development lifecycle phase, their management system requirements based on their scope of work will be defined in contractual agreements.

### 17.3 Quality Management

Within the OPG Nuclear Management System there are several programs that support the quality management of projects such as the design and construction of a new nuclear facility. These programs include OPG-PROG-0039 (Reference 17.6-23), N-PROG-MP-0009 (Reference 17.6-10), and OPG-PROG-0009 (Reference 17.6-21).

OPG-PROG-0039 (Reference 17.6-23), as described in Section 17.2, establishes how the requirements, objectives, and commitments for the management of projects are fulfilled. Such project requirements include the need to follow an appropriate quality program, and that safety and quality take higher priority and are not compromised over schedule or cost. Under this program, any project work performed by a contractor shall be done under an OPG evaluated quality program that meets specified requirements. This may be a singular or multiple documents (manuals, plans, procedures, etc.) that include how they will manage quality through the lifecycle of the contract, how they will apply informed customer principles to procured services and pass quality requirements through to suppliers. The planned measures must also describe controls in their own management system required to execute the contract.

Design activities for a nuclear facility are governed by N-PROG-MP-0009 (Reference 17.6-10), either through full utilization of this program, or in accordance with a contractor quality program approved by OPG. OPG is required to monitor and assess design activities to confirm compliance with procedures.

A contractor's quality program must meet requirements such as CSA N299 (Reference 17.6-37), applicable elements of CSA N286 (Reference 17.6-1), or other appropriate quality standards for the contracted work as stated in contractual agreements. As per the OPG-PROG-0009 (Reference 17.6-21) a contractor's qualification is established based on codes, standards and design and quality requirements. OPG conducts contractor prequalification assessments to evaluate this quality program and performs qualification audits to verify that the contractor's management system meets requirements of CSA N286 (Reference 17.6-1) and other QA requirements as required. The deliverables identified in a contract will be completed under the contractor's programs, and then reviewed and accepted by OPG. OPG applies an informed customer approach. Utilizing this approach, OPG is responsible to clearly define requirements for the procured items/services and control of the supply chain leading to work completion, to monitor work execution and oversee the quality of subcontracted work, including technical review of the work performed by contractors during and after work was completed. Such an oversight is accomplished by a set of processes which includes:

- Supplier evaluation and selection
- Supplier monitoring and Quality Surveillance
- Evaluation and release of final product

Contractors will provide quality control per their quality plans and will have processes to address identified non-conformances. OPG will provide Quality Surveillance ensuring contractor compliance with their quality plans. Quality Surveillance provides assurance that contractor activities meet the specified requirements of the accepted quality standard requirement and OPG's Nuclear Management System.

### **17.3.1 Facility Development Lifecycle Phases and Activities**

#### **17.3.1.1 OPG Quality Management**

Details of how OPG's quality management processes are implemented in the Nuclear Management System are described above in Section 17.3. OPG as the owner and licensee is accountable for the quality program framework and governance that assures compliance with the OPG Nuclear Management System, and for the oversight of design, procurement, construction, and commissioning activities for the DNNP. OPG's oversight activities of a contractor's quality program will ensure that regulatory and contractual requirements are met and that the work is performed by qualified staff.

For the development lifecycle phase, the DNNP shall be implemented using a graded approach commensurate with risk in application of QA requirements as defined in CSA N286 (Reference 17.6-1).

Utilizing OPG's experience with large projects such as the Darlington Refurbishment, the Quality Surveillance process will be implemented using a Systematic Approach to Project Surveillance. This process will be implemented on a risk and sampling basis to address quality and regulatory requirements.

As the project progresses, details on how DNNP will meet the intent of OPG's Nuclear Management System and description of program-specific requirements, including quality management, are introduced in NK054-PLAN-01210-00008 (Reference 17.6-27).

As stated in Section 17.3, any organization outside of OPG contracted to perform project work, such as development lifecycle activities for the DNNP, is required to meet specified quality standard requirements as defined in associated contractual agreements. These are evaluated in accordance with procurement activities required by OPG-PROG-0009 (Reference 17.6-21). As GEH is currently under contract with OPG to perform development lifecycle activities and is the primary contributor of design of the DNNP, a description of the GEH Quality Management System is provided in Subsection 17.3.1.2.

#### **17.3.1.2 GEH Quality Management**

GEH has an established and robust design management and control process that is part of the Integrated Management System described in Section 17.2.1.2 that will be utilized for the development of the BWRX-300 Facility. The design of the BWRX-300 for DNNP will be implemented, using a graded approach, through the following project specific plans:

- DNNP Quality Management Plan
- DNNP Design Management Plan
- DNNP Project Management Plan

The above plans, along with the GEH management system, work together to ensure all requirements of the BWRX-300 facility at the DNNP site are safely and economically implemented in a timely manner.

#### **Quality Management Plan**

GEH activities performed for the DNNP BWRX-300 facility are controlled in accordance with a quality management plan. In the context of the quality management plan, the USNRC approved QAPD is referred to as the GEH Quality Assurance Program.

The quality management plan supplements the QAPD to assure that GEH activities performed for the DNNP comply with project specific (i.e., OPG and CNSC) quality requirements.

The GEH QAPD has been evaluated to CSA N286-12, CSA N286.7, and CSA N299.1-16. Identified gaps are addressed through the implementation of the quality management plan.

The selection of quality requirements for the BWRX-300 facility is governed by the quality management plan.

### **Design Management Plan**

GEH design activities are conducted and controlled in accordance with a design management plan.

The key elements of the Design Management plan that are important to the management of safety are implemented in accordance with supporting plans specific to:

- Design authority
- Requirements Management
- SSC Classification
- Research and Development
- Configuration Management
- Information Management

The above supporting plans are further described below.

### **Design Authority**

The prime responsibility for safety of the facility rests with OPG which is the owner, Operator, and licensee. This responsibility includes design activities performed by OPG and the oversight of design activities performed by contracted organizations, such as GEH.

The Design Authority is the entity that has overall responsibility for the management of design including approving design changes, defining and controlling the design basis, maintaining configuration control, ensuring work is performed by qualified and competent people/organizations, and ensuring that the requisite knowledge is maintained. OPG has identified GEH as Design Authority for scope of work defined by contractual arrangements.

### **Requirements Management**

The approach for managing the flow down of the plant requirements to the system level and respective component level requirements.

Requirements Management tracks the requirements and their attributes from identification to validation and testing. The following are the main objectives of Requirements Management:

Traceability: Establish relationship between requirements and verification and validation.

Change Impact Assessment: Facilitate determination of impact on upstream and downstream requirements when design changes are proposed.

Completeness of Design: Provide confidence that stakeholder requirements are fulfilled in the body of design documentation. This aspect of Requirements Management is sometimes referred to as “forward traceability,” which results in an auditable trail of requirements in the design documentation and evidence that the completed design meets the requirements.

Control of Scope: Help prevent introduction of functions and features that are not required by imposing discipline during the design process such that requirements are not introduced without a basis. This aspect of Requirements Management (RM) is sometimes referred to as “backward traceability.”

Consistency: A defined RM strategy helps ensure that various organizations and disciplines have a common understanding about how requirements are managed, how the requirements architecture is used, and how the RM tools are used.

### **Structures, Systems, and Components Classification**

The BWRX-300 approach to classifying SSC is based on the classification principles contained in IAEA SSR-2/1, “Safety of Nuclear Power Plants: Design” (Reference 17.6-39) and IAEA SSG-30, “Safety Classification of Structures, Systems, and Components in Nuclear Power Plants” (Reference 17.6-40).

See Chapter 3, Section 3.2 for details of the BWRX-300 SSC Classification Methodology.

### **Research and Development**

Research and Development activities are those that support the development and deployment of the BWRX-300.

The BWRX-300 is an evolution of the USNRC certified Economic Simplified Boiling Water Reactor (ESBWR), and existing Boiling Water Reactors (BWR). Operating experience for the fleet of BWRs, Advanced Boiling Water Reactors (ABWR), ESBWR licensing, and decommissioning of BWRs are applied to the BWRX-300 design.

The BWRX-300 Research and Development program applies to all phases of the BWRX-300 lifecycle. The use of Research and Development activities and operating experience to shape BWRX-300 design, testing, and operations decisions is an effort that is not expected to end until all related units have been decommissioned.

The Research and Development program is part of the design process that is part of the GEH management system; however, not all activities in the Research and Development program are considered design activities. The GEH Quality Assurance Program requirements are applied to Research and Development program activities commensurate with the importance of the activity to safety.

### **Configuration Management**

Configuration Management is established and maintained to ensure the design basis of the facility is effectively managed throughout the development of the BWRX-300 Power Block at the DNNP site. It is a key element that ensures the responsibilities of the Design Authority role are sufficiently performed.

An important objective of Configuration Management is to ensure that accurate and current information is available in a timely manner to allow a responsible individual to make safe, knowledgeable, and cost-effective decisions.

NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

American National Standards Institute/Nuclear Information and Records Management Association (ANSI/NIRMA) CM 1.0, "American National Standard Guidelines for Configuration Management for Nuclear Facilities" (Reference 17.6-41), defines Configuration Management as the systematic approach for identifying, documenting, and changing the characteristics of a Plant's SSC and ensuring that conformance is maintained between the Design Requirements, Physical Configuration, and Facility Configuration Information. This concept is illustrated in Figure 17.3-1 showing that the Configuration Management equilibrium is established when, "what needs to be there" conforms to "what we say is there" conforms to "what is actually there."

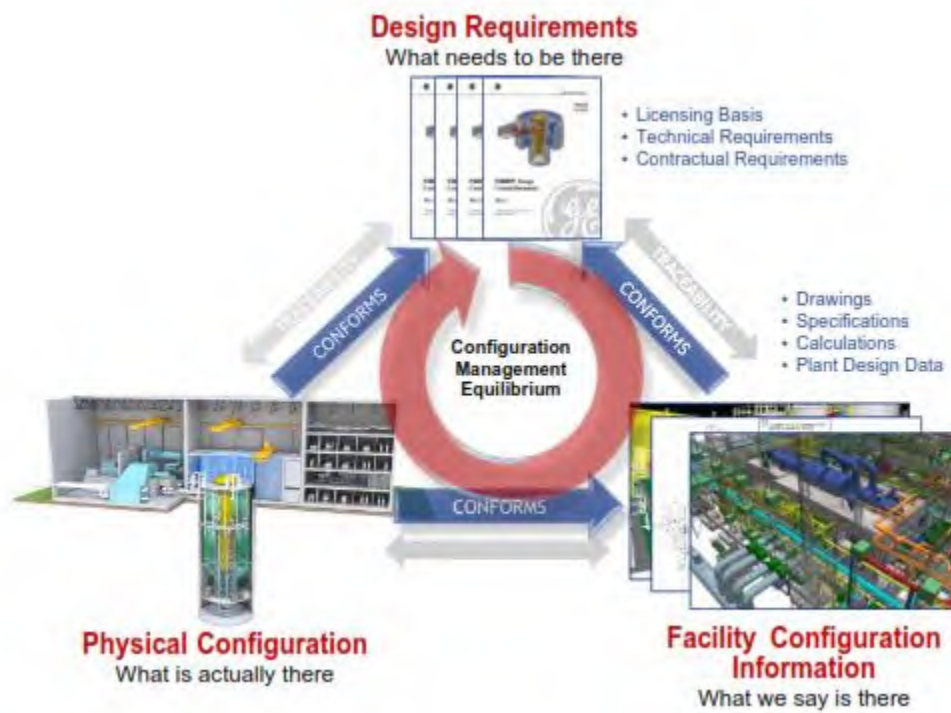


Figure 17.3-1: Configuration Management Model



## **Information Management**

Information management provides the framework, strategy, and implementation plan to deploy and support the overall setup and configuration of the Content Management System for the development of the BWRX-300 at the DNNP site.

Information Management, Design Requirements and Configuration Management work together to ensure Configuration Management equilibrium is achieved.

## **Project Management**

A Project Management Organization is established to plan, execute, control, and monitor work activities; drive conceptual design and estimating objectives; and ensure work is performed in accordance with project requirements. For the purposes of the QAPD, the Project Management Organization is the Operating Organization for the OPG project.

The project work plan is further supported by additional sub-ordinate tools that are key to the management of safety, including but not limited to:

- Project procurement
- Risk management

## **Project Procurement**

Project procurement supports GEH procurement activities performed during the development of the BWRX-300 at the DNNP site.

Its purpose is to provide programmatic instructions and non-technical guidance to GEH team members for preparation and issuance of purchase and construction specifications, development of procurement packages for requests for proposals and request for quotations to potential suppliers, bid evaluations, and contract administration with the selected supplier(s) until delivery of equipment to Darlington OPG Site using established processes for the BWRX-300 project application.

The procurement process is adequately evaluated and reviewed throughout the project lifecycle to ensure the project includes proper strategies, communication, planning, decision making, and timely implementation of project requirements leading to the success of the overall project.

## **Risk Management**

Project risk management describes how the risk management process will be structured and performed for the development of the BWRX-300 and its deployment at the DNNP site.

Risk Management provides a formal process that follows the Project Management Book of Knowledge Guide that will help reduce the overall project risk exposure and increase the likeliness of achieving the project's requirements, cost, and schedule objectives. Along with documenting the process of identifying, assessing, treating, and monitoring risks, key personnel, key performance indicators, review update frequencies, and reporting requirements.

### **17.3.1.3 Constructor and Architect Engineering Firm Quality Management**

The Constructor and Architect Engineering Firm will be required to meet specified quality standard requirements based on their scope of work as defined in associated contractual agreements.

Oversight of such requirements by OPG, will be performed through the processes outlined in the Items and Services Management Program, OPG-PROG-0009 (Reference 17.6-21).

### **17.3.2 Facility Operational Lifecycle Phases and Activities**

The DNNP that will transition from a project to a nuclear facility during the operational lifecycle phase will require operational programmatic governance to be in place prior to use. This transition may warrant changes to the quality management processes. Any required changes to associated governance to support this lifecycle phase will be analyzed as part of the Licence to Operate application process.

### **17.3.3 Facility End of Life Lifecycle Phases and Activities**

The quality management processes utilized by the future DNNP facility may evolve over the lifetime of the facility and continue to meet regulatory requirements. Any required changes to related governance to support this lifecycle phase will be analyzed to support the Licence to Decommission application process.

## **17.4 Measurement, Assessment, and Improvement of the Owner Management System**

OPG is the owner and the licence applicant for the DNNP project. The OPG Chief Nuclear Officer is accountable for the effectiveness of the overall Nuclear Management System and ensuring compliance with the requirements set out in CSA N286 (Reference 17.6-1). The following lists applicable programs given authority from N-CHAR-AS-0002 (Reference 17.6-3) that support the measurement, assessment, and continual improvement of the Nuclear Management System (descriptions of these programs can be found in Section 17.2).

- N-PROG-AS-0001 (Reference 17.6-8)
- N-PROG-RA-0010 (Reference 17.6-15)
- N-PROG-RA-0003 (Reference 17.6-14)

### **17.4.1 Facility Development Lifecycle Phases and Activities**

#### **17.4.1.1 OPG Management, Assessment, and Improvement of the Management System**

Details of how OPG measures, assesses, and improves its Nuclear Management System are described in Section 17.4. For the development lifecycle phase, the DNNP will utilize these programs. As these programs primarily support the Nuclear Management System effectiveness at the program or fleet/site level, the DNNP, will utilize a subset of the Nuclear Management System programs as described in Section 17.2, to the extent required of a project not performing any nuclear activities during this lifecycle phase.

Any documentation that may be produced to support the contract partners for the development lifecycle phase, that could have potential to benefit the operational lifecycle, will be analyzed for inclusion into the programs and/or processes to be utilized by the future nuclear facility as part of the Licence to Operate application process.

As GEH is currently under contract with OPG to perform development lifecycle activities for the DNNP, a description of the GEH performance improvement elements is provided in Subsection 17.4.1.2.

#### **17.4.1.2 GEH Performance Improvement Elements**

Performance Improvement is an integral element of the GEH management system.

As described in Subsection 17.2.1.2 above, Section 16 of the QAPD (further described below) establishes the requirements to establish the necessary measures to promptly identify, control, document, classify, and correct Conditions Adverse to Quality.

The GEH Corrective Action Program utilizes a system of implementing procedures to establish the necessary measures to promptly identify, control, document, classify, and correct Conditions Adverse to Quality.

Implementing procedures for the Corrective Action Program ensure that:

1. Corrective actions from audits and self-assessments, and non-conformances are analyzed to detect conditions or potential Conditions Adverse to Quality
2. Changes in processes or procedures resulting from corrective and preventive actions are implemented and recorded

The Corrective Action Program and related tools are utilized to document, track status, and maintain records of corrective and preventive action activities and results.

Reports of Conditions Adverse to Quality are analyzed to identify trends in quality performance.

Significant conditions and trends adverse to quality are reported to the appropriate level of management.

As described in Subsection 17.2.1.2 above, Section 18 of the QAPD establishes the requirements and processes for a comprehensive audit program to verify implementation of and compliance to all aspects of the quality system including its adequacy and effectiveness. The audit program requirements apply to activities affecting the quality of services, hardware, software, and personnel who perform related activities or may affect quality. Project surveillances are planned, executed, and recorded to evaluate through investigation or examination, the adequacy of activities (design, manufacturing, quality processes, etc.) and the compliance to established procedures, instructions, and other applicable documents.

GEH conducts annual reviews of its management system to ensure its suitability, adequacy, and effectiveness. The review team assesses opportunities for improvement and the need for changes to the quality system, including quality policy and quality objectives.

The performance improvement elements described above are implemented through established processes and provide for a continuous evaluation of the GEH management system to achieve desired business performance.

#### **17.4.1.3 Constructor and Architect Engineering Firm Performance Improvement Elements**

Additional organizations outside of OPG or GEH contracted to perform development lifecycle activities the DNNP, such as the Constructor and Architect Engineering Firm, will be required to meet specified requirements for the measurement, assessment and improvement of their management system based on their scope of work as defined in associated contractual agreements.

Oversight of such requirements by OPG, will be performed through the processes outlined in the OPG-PROG-0009 (Reference 17.6-10).

#### **17.4.2 Facility Operational Lifecycle Phases and Activities**

The DNNP that will transition from a project to a nuclear facility during the operational lifecycle phase, will require operational programmatic governance to be in place prior to use. This transition must maintain the requirement for the measurement, assessment, and improvement of the management system to be in place for the operating life of the facility. Any required changes to associated governance to support this lifecycle phase will be analyzed as part of the Licence to Operate application process.

#### **17.4.3 Facility End of Life Lifecycle Phases and Activities**

The programs and associated processes involved with the measurement, assessment, and improvement of the Nuclear Management System utilized by the future DNNP facility may evolve over the lifetime of the facility and continue to meet regulatory requirements. Any required changes to associated governance to support this lifecycle phase will be analyzed to support of the Licence to Decommission application process.

## **17.5 Fostering a Culture for Safety**

As required by CSA N286 (Reference 17.6-1), and in compliance with CSNA REGDOC-2.1.2, "Safety Culture" (Reference 17.6-42), a management system shall be used to understand and promote a safety culture in every phase of a nuclear facility lifecycle. N-POL-0001 documents OPG's commitment to fostering a safety and security culture and gives authority to the Nuclear Management System. The policy states that:

"Nuclear Safety and Security shall be the overriding priority in all activities performed in support of OPG nuclear facilities. Nuclear Safety shall have clear priority over schedule, cost, and production."

As defined in the policy, the Chief Nuclear Officer is accountable to the Chief Executive Officer and the Board of Directors to establish a management system that fosters nuclear safety and security as the overriding priority. In addition, OPG commits to conducting comprehensive, systematic, and rigorous safety culture assessments at least every 5 years. The policy also makes everyone accountable to conduct themselves in a manner consistent with the following nuclear safety & security culture traits:

- Personal Accountability
- Questioning Attitude
- Effective Safety Communication
- Leadership Safety Values and Actions
- Decision Making
- Respectful Work Environment
- Continuous Learning
- Problem Identification and Resolution
- Environment for Raising Concerns
- Work Processes
- Vigilance

Receiving authority from this policy, N-CHAR-AS-0002 (Reference 17.6-3) defines the framework that establishes the programs and processes required to ensure OPG achieves all safety objectives and continuously monitors performance against these objectives. The charter describes that every employee that executes or supports licensed activities and every person or entity that supplies a product or service for OPG's nuclear facilities is responsible and shall be held accountable for complying with the expectations within the charter and programs that make up the Nuclear Management System. This includes that contract partners demonstrate the attributes of a positive safety culture. The following Nuclear Management System programs further support the implementation and continuous improvement of OPG's safety and security culture (descriptions of these programs can be found in Section 17.2).

- N-PROG-AS-0001
- OPG-PROG-0005
- N-PROG-AS-0002
- N-PROG-RA-0003

## **17.5.1 Facility Development Lifecycle Phases and Activities**

### **17.5.1.1 OPG Safety Culture Elements**

Details of how the OPG nuclear safety and security culture is defined and implemented in the Nuclear Management System are described above in Section 17.5. The DNNP will utilize this Nuclear Management System governance for the development lifecycle phase.

Any organization outside of OPG contracted to perform development lifecycle activities is required to meet the same CSA N286-12 requirement to understand and promote a safety culture. As GEH is currently under contract with OPG to perform development lifecycle activities for the DNNP, a description of the GEH safety culture elements is described in Subsection 17.5.1.2.

### **17.5.1.2 GEH Safety Culture Elements**

Safety Culture is at the core of the GEH organization and is an integral element of its management system being implemented through its highest tier defining document “Nuclear Safety & Security Culture Policy” and Part III of its QAPD.

A key nuclear safety and security culture trait is a safety conscious work environment, which is a work environment where individuals feel free to raise nuclear safety concerns without fear of retaliation, intimidation, harassment, or discrimination. It is GEH’s policy to foster such an environment by encouraging employees to raise nuclear safety and security concerns; providing alternate reporting mechanisms through which those concerns may be raised (e.g., supervision, Corrective Action Program, ombuds program); and, commensurate with their potential nuclear safety and security significance, promptly reviewing, prioritizing, and resolving the concerns with timely feedback to the originator.

Values are premised on the principle that everyone, regardless of position, is responsible for nuclear safety and security and that their behaviors must reflect a strong questioning attitude, conservative decision making, and safety-over-output prioritization.

GEH management performs the following actions to establish the appropriate safety conscious work environment:

1. Ensures common understanding of the key aspects of safety culture within the organization. Provides the means by which the organization supports individuals and teams in carrying out their tasks safely and successfully, taking into account the interaction between individuals, technology and the organization.
2. Reinforces a learning and questioning attitude at all levels of the organization.
3. Provides the means by which the organization continually seeks to develop and improve its safety culture.
4. Provides the means by which the organization assess the performance of its safety culture.

### **17.5.1.3 Constructor and Architect Engineering Firm Safety Culture Elements**

Additional organizations outside of OPG or GEH contracted to perform development lifecycle activities for the DNNP, such as the Constructor and Architect Engineering Firm will be required to meet the same CSA N286(Reference 17.6-1) requirement to understand and promote a safety culture as applicable based on their scope of work. This requirement will be specified, if required, in the associated contractual agreement.

Oversight of such requirements by OPG, will be performed through the processes outlined in OPG-PROG-0009 (Reference 17.6-21).

### **17.5.2 Facility Operational Lifecycle Phases and Activities**

The DNNP that will transition from a project to a nuclear facility during the operational lifecycle phase, will require operational programmatic governance to be in place prior to use. This transition must maintain the requirements for safety culture to be in place for the operating life of the facility. Any required changes to associated governance to support this lifecycle phase will be analyzed as part of the Licence to Operate application process.

### **17.5.3 Facility End of Life Lifecycle Phases and Activities**

The safety culture of the future DNNP facility may evolve over the lifetime of the facility and continue to meet regulatory requirements. Any required changes to related governance to support this lifecycle phase will be analyzed to support the Licence to Decommission application process.

## **17.6 References**

- 17.6-1 CSA N286, "Management System Requirements for Nuclear Facilities," CSA Group.
- 17.6-2 CNSC Regulatory Document REGDOC-2.1.1, "Management System."
- 17.6-3 N-CHAR-AS-0002, "Nuclear Management System Charter," Ontario Power Generation.
- 17.6-4 OPG-POL-0032, "Safe Operations Policy," Ontario Power Generation.
- 17.6-5 N-POL-0001, "Nuclear Safety & Security Policy," Ontario Power Generation.
- 17.6-6 PRSL 18.00/2031, "Power Reactor Site Preparation Licence," Ontario Power Generation.
- 17.6-7 LCH-DNNP-PRSL 18.00 2031, "Licence Condition Handbook," Ontario Power Generation.
- 17.6-8 N-PROG-AS-0001, "Nuclear Management System Administration," Ontario Power Generation.
- 17.6-9 N-PROG-AS-0002, "Human Performance," Ontario Power Generation.
- 17.6-10 N-PROG-MP-0009, "Design Management," Ontario Power Generation.
- 17.6-11 N-PROG-MP-0014, "Reactor Safety Program," Ontario Power Generation.
- 17.6-12 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 17.6-13 N-PROG-RA-0002, "Conduct of Regulatory Affairs," Ontario Power Generation.
- 17.6-14 N-PROG-RA-0003, "Performance Improvement," Ontario Power Generation.
- 17.6-15 N-PROG-RA-0010, "Independent Assessment," Ontario Power Generation.
- 17.6-16 N-PROG-RA-0011, "Nuclear Security," Ontario Power Generation.
- 17.6-17 N-PROG-RA-0015, "Safeguards and Nuclear Material Accountancy," Ontario Power Generation.
- 17.6-18 N-PROG-TR-0005, "Training," Ontario Power Generation.
- 17.6-19 OPG-PROG-0001, "Information Management," Ontario Power Generation.
- 17.6-20 OPG-PROG-0005, "Environment Health and Safety Managed Systems," Ontario Power Generation.
- 17.6-21 OPG-PROG-0009, "Items and Services Management," Ontario Power Generation.
- 17.6-22 OPG-PROG-0037, "OPG Business Planning," Ontario Power Generation.
- 17.6-23 OPG-PROG-0039, "Project Management," Ontario Power Generation.
- 17.6-24 OPG-PROG-0042, "Cyber Security," Ontario Power Generation.
- 17.6-25 W-PROG-WM-0003, "Decommissioning Program," Ontario Power Generation.
- 17.6-26 NK054-REP-08130-00004, "DNNP Licence to Construct Management System Report," Ontario Power Generation.
- 17.6-27 NK054-PLAN-01210-00008, "Darlington New Nuclear Project – Program Management Plan," Ontario Power Generation.



NEDO-33967 REVISION 0  
NON-PROPRIETARY INFORMATION

- 17.6-28 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 17.6-29 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- 17.6-30 10 CFR 71 Subpart H, "Quality Assurance."
- 17.6-31 10 CFR 21, "Reporting of Defects and Noncompliance."
- 17.6-32 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 17.6-33 ISO 9001, "Quality Management Systems - Requirements," International Organization for Standardization.
- 17.6-34 IAEA GSR Part 2, "Leadership and Management for Safety," International Atomic Energy Agency.
- 17.6-35 KTA 1401, "General Requirements for the Quality Assurance," Nuclear Safety Standards Commission.
- 17.6-36 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 17.6-37 CSA N299.1, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 1," CSA Group.
- 17.6-38 CNSC Regulatory Document REGDOC-2.4.3, "Nuclear Criticality Safety." International Atomic Energy Agency.
- 17.6-39 IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design," International Atomic Energy Agency.
- 17.6-40 IAEA SSG-30, "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants,"
- 17.6-41 ANSI/NIRMA CM 1.0, "American National Standard Guidelines for Configuration Management for Nuclear Facilities," American National Standards Institute.
- 17.6-42 CNSC Regulatory Document REGDOC-2.1.2, "Safety Culture."



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33968

Revision 0

September 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 18  
Human Factors Engineering**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licencing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
AOF	Allocation of Functions
CBP	Computer-Based Procedures
CNSC	Canadian Nuclear Safety Commission
COO	Concept of Operations
COTS	Commercial-Off-The-Shelf
DCT	Data Connection Table
DNNP	Darlington New Nuclear Project
DSA	Deterministic Safety Analysis
EOP	Emergency Operating Procedure
FRA	Functional Requirements Analysis
GEH	GE-Hitachi Nuclear Energy
HED	Human Engineering Discrepancy
HF	Human Factors
HFE	Human Factors Engineering
HFEITS	Human Factors Engineering Issue Tracking System
HFEPP	Human Factors Engineering Program Plan
HPM	Human Performance Monitoring
HRA	Human Reliability Analysis
HSI	Human-System Interface
I&C	Instrumentation and Control
ISV	Integrated System Validation
MCR	Main Control Room
NUREG	Nuclear Regulatory Report
OE	Operating Experience
OER	Operating Experience Review
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
SAA	Severe Accident Analysis
SCR	Secondary Control Room
SPDS	Safety Parameter Display System
SSC	Structures, Systems, and Components

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

Acronym	Explanation
T&E	Testing and Evaluation
TA	Task Analysis
TSV	Task Support Verification
UIS	User Interface Specification
V&V	Verification and Validation

## TABLE OF CONTENTS

<b>18.0 HUMAN FACTORS ENGINEERING .....</b>	<b>18-1</b>
18.1 Management of the Human Factors Engineering Program .....	18-1
18.1.1 HFE Program Goals .....	18-1
18.1.2 Program Scope.....	18-2
18.1.3 Team and Organization .....	18-9
18.1.4 Process and Procedures.....	18-12
18.1.5 Issue Tracking .....	18-14
18.1.6 References .....	18-15
18.2 Human Factors Engineering Analysis .....	18-16
18.2.1 Review of Operating Experience.....	18-16
18.2.2 Functional Requirements Analysis.....	18-17
18.2.3 Allocation of Function .....	18-18
18.2.4 Task Analysis .....	18-20
18.2.5 Staffing .....	18-23
18.2.6 Treatment of Important Human Actions .....	18-24
18.2.7 References .....	18-26
18.3 Design of the Human-System Interface .....	18-27
18.3.1 Design Goals and Design Bases .....	18-27
18.3.2 Human-System Interface: Design Inputs.....	18-28
18.3.3 Human-System Interface: Detailed Design and Integration.....	18-29
18.3.4 Human-System Interface: Tests and Evaluations .....	18-32
18.3.5 Human-System Interface: Design of the Main Control Room .....	18-35
18.3.6 Human-System Interface: Design of the Secondary Control Room.....	18-37
18.3.7 Procedure Development .....	18-38
18.3.8 Training and Qualification Program Development.....	18-40
18.3.9 References .....	18-43
18.4 Human Factors Verification and Validation .....	18-45
18.4.1 Objectives and Scope.....	18-45
18.4.2 Methodology .....	18-46
18.4.3 Results .....	18-48
18.4.4 References .....	18-48
18.5 Design Implementation.....	18-49
18.5.1 Objectives and Scope.....	18-49

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

18.5.2	Methodology .....	18-49
18.5.3	Results .....	18-49
18.6	Human Performance Monitoring .....	18-50
18.6.1	Objectives and Scope .....	18-50
18.6.2	Methodology .....	18-50
18.6.3	Results .....	18-51



**LIST OF TABLES**

Table 18.1-1: Mapping of HFE Program Elements with CNSC and CSA Requirements.....18-6

**LIST OF FIGURES**

Figure 18.4-1: BWRX-300 HFE V&V Process Overview .....	18-46
--	-------

## **18.0 HUMAN FACTORS ENGINEERING**

Preliminary Safety Analysis Report (PSAR) Chapter 18 presents the Human Factors Engineering (HFE) program for the BWRX-300 to demonstrate the adequacy of integration of HFE requirements and analysis results into the plant design. The program of HFE activities and analysis informing the design of the plant Structures, Systems, and Components (SSC) is based on clear definition of the full plant set of users and a clearly defined scope of application across the full plant design, operational modes, and lifecycle stages, with focus on important human actions. The HFE content for this PSAR chapter reflects the level of maturity of the HFE Program, plant design, and safety analyses at the time of submission. The pre-operational safety analysis report details further design and analyses development and summarizes HFE Program progression in support of the Licence to Operate submission.

Chapter 18 provides a summary of the BWRX-300 Human-System Interface (HSI) design goals and bases, analyses undertaken to understand the plant-specific HFE requirements related to task performance, the process for detailed HSI design, and activities supporting effective design implementation. The overall design and implementation process is described in detail in NEDC-33982P, "BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan" (Reference 18.1-1). The Human Factors Engineering Program Plan (HFEPP) presents the comprehensive, iterative design approach used for the development of human-centred interfaces and work environment for the plant.

Note that Section 18.1 provides an overview of the HFE Program and outlines its activities or "technical elements". The remainder of the chapter provides the details, including the scope and summary of methods, of the technical elements outlined in Section 18.1.

### **18.1 Management of the Human Factors Engineering Program**

#### **18.1.1 HFE Program Goals**

The high-level goal of the BWRX-300 HFE Program is to conduct a proportionate, integrated, and effective set of HFE design activities that considers users and all phases of the plant lifecycle and that result in a design that reduces the risks and consequences related to human interactions with the plant to as low as reasonably achievable. The program was developed and conducted in line with multiple nuclear regulatory requirements, particularly those in Canadian Nuclear Safety Commission (CNSC) REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility" (Reference 18.1-2), CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.1-3) and CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.1-4).

The HFEPP defines the program and outlines the way that the specified general human-centred HFE design goals are operationalized and verified during the design process. This is achieved, through the application of the HFE analyses, integrated design and safety analysis support, and provision of tools, technical requirements, and guidance to designers. The HFE Program ensures that the plant-level design goals are achieved, including:

1. Design of HSIs reduces the likelihood of error and provides for timely, clear error detection.
2. Tasks can be accomplished within time and performance criteria.
3. Allocation of Function (AOF) and proposed job design (staff complement and job roles) are such that a suitable level of human vigilance is ensured and acceptable workload levels that minimize periods of human underload and overload is provided.
4. Presentation of information supports a high degree of situational awareness of the state of the plant and actions required.

5. HSI design supports the capability of personnel to recover from previous decisions and actions that did not achieve intended results.
6. Application of ergonomic principles to working areas and their environments ensure these areas are safe and designed to support performance of required tasks.

The above HFE goals are embedded into the design through specification of requirements derived from codes, standards, best-practice guidance, and plant-specific HFE analyses, and through integrated HFE team support to the design. Achievement of the goals is confirmed using the design tools, HFE Testing and Evaluation (T&E) throughout the design development, and HFE Verification and Validation (V&V) of the realized design.

### **18.1.2 Program Scope**

The HFE Program scope applies to HSI components and SSC within or that form part of facilities, systems, equipment, and components throughout the plant. HSIs are defined as any region or point at which a person interacts with a system, equipment, or component. System interface means any digital and electronic Instrumentation and Control (I&C) user interfaces, as well as hardware-based user interfaces and design features on panels, equipment, and individual components. This includes HSIs within or forming part of:

- Control facilities for reactor operations
- Facilities for supporting response to accidents and emergencies
- Control room or stations for radwaste processing
- Control room or stations supporting refuelling and maintenance outage work
- Local control stations
- Equipment- and process line-mounted HSIs
- Auxiliary and support facilities and equipment located external to the main reactor and powerhouse buildings

This includes specification to and oversight of HSIs that form part of SSC supplied by external vendors, ensuring that supplied design or selection of standard equipment and components is consistent with the HFE requirements of the HFE Program.

The HFE Program also applies across the full scope of users and activities that support plant operation, testing, inspection, and maintenance, including functions such as fuel handling, chemistry, radioactive waste processing, and radiation protection.

The HFE Program described in this plan applies to design activities that consider Human Factors (HF) risks that might arise in all phases of the plant lifecycle, including:

- Construction
- Commissioning
- Operation
- Decommissioning

The HFE Program applies to all HSIs, including those at the following locations:

- Main Control Room (MCR)
- Secondary Control Room (SCR)

- Emergency Response and Support Facilities
- Radwaste Building Control Room or Control Stations
- Local Control Station interfaces
- Equipment- and process line-mounted interfaces (e.g., control actuators and gauges)
- HSIs related to auxiliary and support facilities located outside of the main buildings (e.g., hydrogen tanks or fuel oil supplies)

The HFE Program applies to all plant conditions in the design basis, including normal, outage (refuelling and maintenance outages, including extended refurbishments), abnormal, emergency, and accident conditions. The scope of the HFE Program is extensive; however, the application of HFE support and activities to the scope of each phase and task location is graded (or proportionate), as discussed in Subsection 18.1.4.2 to apply a higher level of emphasis and rigour for important human interactions that are safety-critical or hazardous.

Note that for some phases of the plant lifecycle, particularly Construction and Decommissioning, the HFE activities are focused at a high level. By nature of the single iteration of these plant stages outside of commercial operations, they generally do not involve analysis of recurring operationally related functions and tasks. HFE in design related to these non-commercial operational phases is centred around providing basic guidelines and ensuring design strategies that aid in the achievability of the overall goals of the phase. For example, for decommissioning, HFE design guidelines and requirements for maintainability may equally apply, especially clearance and access for removal of large components and equipment, and consideration of radiological safety through plant structures and equipment layout.

The same general HFE methodologies and tools described in this chapter are also applied to the HSIs related to security. A risk-based approach to HFE design requirements, task support requirements, and testing methodologies is also applied to security considerations. However, due to the sensitive nature of the specific details of security risk ratings, credited human actions, security success criteria, and testing scenarios, the HFE activities for security are found within the Security Annex.

The HFEPP at the time of this PSAR, and in support of the Licence to Construct application, is focused on the design of the plant. After plant turnover to the utility, the utility HFE Program is defined through suitable processes and procedures to address HFE licenced and operating plant activities such as Human Performance Monitoring (HPM), management of change for operational documentation, and design modification activities.

#### **18.1.2.1 Overview of the Human Factors Engineering Program**

The HFEPP describes the goals and scope of the HFE Program, along with items such as:

1. Assumptions and constraints in conducting the program
2. Coordination of the HFE Program with the overall plant design activities, including coordination with the plant safety analysis
3. Tools and facilities (e.g., mock-ups, computer simulations) used in support of the program
4. Composition, qualifications, and responsibilities of the HFE organization
5. Process and procedures followed including the process for identifying and managing technical and programmatic issues
6. Documentation developed

7. Summary of how the results of the HFE analysis are incorporated into the design, operational documentation, and safety analyses

The HFEPP defines each of the technical elements, the specific activities that comprise the full integrated program, as outlined in the next section.

#### **18.1.2.2 Human Factors Engineering Program Technical Elements**

The technical elements for the HFE Program are described briefly below. The full description of these elements, and how they constitute a comprehensive and robust program of HFE integration across the plant design, forms the remainder of this chapter (Sections 18.2 through 18.6).

1. *Operating Experience Review (OER)* – identification, review and incorporation of any recommendations and learning (positive and negative) from past events and user feedback related to HFE in design (Subsection 18.2.1)
2. *Functional Requirements Analysis (FRA)* – determination of functions required to achieve plant goals in all plant states (Subsection 18.2.2)
3. *Allocation of Function (AOF)* – assigning the identified functions to system (technology) or human, based on respective capabilities and limitations of each (Subsection 18.2.3)
4. *Task Analysis (TA)* – identification of the tasks required to achieve the allocated functions, and decomposition into task steps to allow the identification and characterization of HSIs, personnel, locations, and support equipment (e.g., communications, lighting, personnel protection) required to perform each task successfully (Subsection 18.2.4)
5. *Staffing Analysis* – determination of the numbers and roles of personal required to support optimal task performance in all plant conditions (Subsection 18.2.5)
6. *Treatment of Important Human Actions* – activities supporting and providing input to the BWRX-300 safety analyses to ensure clear identification of human actions important to safety, ensure claimed actions are achievable and identify HSIs requiring the highest level of HFE focus and effort (Subsection 18.2.6)
7. *Human-System Interface (HSI) Design* – identification and management of the set of HFE design requirements from standards, codes, and best-practice guidance, and implementation of those requirements plus results from HFE analyses into the design of HSIs, including integration of HFE team design support; also includes HFE T&E activities (Subsection 18.3.1 through 18.3.6)
8. *Procedures* – process and activities for the development of usable and validated operational documentation, for plant task types (Subsection 18.3.7)
9. *Training and Qualifications* – process and activities for the development of relevant and validated training content, optimized for the plant design, operational documentation, and baseline personnel qualifications and attributes (Subsection 18.3.8)
10. *Human Factors Verification and Validation (V&V)* – detailed, staged set of activities to provide assurance of the correct and sufficient implementation of HFE requirements in the design, and the appropriate design to support required tasks (Section 18.4)
11. *Design Implementation* – support and monitoring of the design from “on paper” to a realized constructed plant, including integration with configuration control to ensure no loss of integrity of the HFE Program goals throughout fabrication and construction (Section 18.5)

12. *Human Performance Monitoring (HPM)* – continuous monitoring of user task performance throughout the lifetime of the plant to ensure optimum plant and organizational design and identify early trends and issues that require HFE design improvements (Section 18.6)

*NOTE: This element is only relevant within the future operational plant HFE Program and therefore not currently included in the HFEPP.*

The scope and nature of the HFE Program, the HFE organization undertaking the program and the technical elements that comprise it, align with regulatory requirements and international standards and guidance, providing assurance that HFE has been suitably and sufficiently integrated into the plant design.

Table 18.1-1 illustrates the alignment of Chapter 18 and the HFE Program technical elements, with the CNSC expectations for HFE, as per CNSC REGDOC-2.5.1 (Reference 18.1-3) and CSA N290.12-14, “Human factors in design for nuclear power plants” (Reference 18.1-5). The PSAR and HFE Program technical elements inform the subsequent pre-operational safety analysis report and ultimately the overall plant lifecycle of HFE activities.

Although arranged somewhat differently in the PSAR chapter and HFE Program, all required elements are included and addressed. The elements are performed in an iterative manner, with activities and outputs progressively evolving with the design and related safety analyses. These elements inform one another, inform, and are informed by plant design and safety analyses, and are aligned with design and engineering processes and requirements, international best-practice guidance, and regulations.

**Table 18.1-1: Mapping of HFE Program Elements with CNSC and CSA Requirements**

ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
--	18.1 Management of the Human Factors Engineering Program	HFEPP, including: Goals, Scope, Background, Criteria for Areas of Consideration Human Factors Input, including: HFE Organization Roles and Responsibilities, Training Needs and Related Groups Methods, including intended tools and technical guides HFE Processes and Procedures Timelines, including logical links to related project activities Documentation Disposition of Human Factors Issues	<i>HF Planning:</i> Determine methods, analyses, evaluations, project interfaces, and tools Identify constraints and drivers Graded approach based on risk and complexity Organization and resources Communications Source documents Issue identification and resolution Documentation Scheduling HFE Interfaces with other groups
--	18.2 Human Factors Engineering Analysis		
1	18.2.1 Review of Operating Experience	Operating Experience Review	<i>HF in Concept Design:</i> OER
2	18.2.2 Functional Requirements Analysis	Functional Analysis	<i>HF in Concept Design:</i> functional analysis
3	18.2.3 Allocation of Function	Allocation of Function	<i>HF in Concept Design:</i> functional analysis (definition includes AOF)
4	18.2.4 Task Analysis	Task Analysis, Job Design	<i>HF in Preliminary Design:</i> TAs including workload and communications analysis; link analysis
5	18.2.5 Staffing	Staffing & Minimum Shift Complement, Job Design, Shift-Work Systems	<i>HF Interfaces:</i> HF in design shall consider the interfaces with staffing; the information common to both HF in design and interfacing disciplines, such as staffing analyses and strategies, should be shared.



NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
6	18.2.6 Treatment of Important Human Actions	Human reliability, activities with potentially hazardous human interactions	<p><i>HF in Concept Design:</i> identification of scenarios to be analyzed</p> <p><i>HF in Preliminary Design:</i> participation in the assessment of human actions and error consequences; assessment of the feasibility of human actions in the deterministic safety analyses</p> <p><i>HF in Detailed Design:</i> confirmation of the feasibility of human actions important to safety in the probabilistic and deterministic safety analyses; analyses to confirm the ability of the human to perform necessary actions</p>
7	18.3 Design of the Human-System Interface 18.3.1 Design Goals and Design Bases	Design human-machine interface system; design physical working environment	
7	18.3.2 Human-System Interface: Design Inputs		<p><i>HF in Concept Design:</i> a statement of system operational purpose and operational requirements under all anticipated conditions; development or selection of HF in design source documents; identification of SSC requirements to support necessary human actions; HFE assessment of design concepts and options</p> <p><i>HF in Preliminary Design:</i> document high-level HF-related requirements; input to specifications and bid evaluations; requirements derived from HF analysis results</p>
7	18.3.3 Human-System Interface: Detailed Design and Integration		<i>HF in Detailed Design:</i> detailed HSI design; design integration of Commercial-Off-The-Shelf (COTS) products

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
7	18.3.4 Human-System Interface: Tests and Evaluations		<i>HF in Preliminary Design:</i> modeling, mock-ups, or prototyping of user interfaces; evaluations <i>HF in Detailed Design:</i> Usability testing
7	18.3.5 Human-System Interface: Design of Main Control Room		Covered within all HF in Design activities
7	18.3.6 Human-System Interface: Design of Secondary Control Room		Covered within all HF in Design activities
8	18.3.7 Procedure Development	Procedure Development	<i>HF in Detailed Design Stage:</i> HF analyses output to development of training manuals, operating procedures, and commissioning procedures
9	18.3.8 Training and Qualification Program Development	Training Program Development	<i>HF in Detailed Design Stage:</i> HF analyses output to development of training manuals, operating procedures, and commissioning procedures
10	18.4 Human Factors Engineering Verification and Validation	Verification Validation	<i>HF in Detailed Design Stage:</i> Verification (carried out before the design is released for construction) Validation (validation activities split between detailed design and implementation)
11	18.5 Design Implementation (post-construction)	Design Implementation	<i>HF in Design Implementation Stage:</i> HFE during installation and commissioning
12	18.6 Human Performance Monitoring (at start of testing)	Human Performance Monitoring	N/A (scope of standard is design stages only)

### 18.1.3 Team and Organization

The HFE team consists of a core and extended team dedicated to integrating HFE requirements and principles into the design. The core HFE team sits within the organization as a separate engineering team, at an equal level to all other discipline teams. The HFE team holds the technical authority over HFE activities and requirements and has the equal authority and issues resolution mechanisms as any other engineering team. The extended HFE team includes members from other disciplines within the engineering organization. This ensures fully integrated and timely consideration of HFE in the daily engineering design decisions and activities.

The core HFE team is comprised of an HFE Technical Lead and two general roles: HF Engineer or HFE Specialist and HFE Operations/Maintenance. The qualifications for these roles may be met by individuals or collectively by the HFE team. The responsibilities and qualifications of these roles are defined in the HFEPP as summarized below:

1. Technical Lead: Provides technical and program oversight and review; responsible for ensuring that HFE activities, interfaces, and outputs meet HFE requirements and align with HFE Program objectives; point-of-contact for schedule development, integration, and management of the program. This role is expected to have the base qualifications of either the HFE Specialist or HFE Operations/Maintenance role, with additional HFE capability across a breadth of HFE competence areas suitable for the full scope of the HFE Program and experience in project management and managing HFE or other technical, cross-cutting programs.
2. HF Engineer/HFE Specialist: Provide specialized knowledge of human cognitive and physical capabilities and limitations, applicable HFE design and evaluation practices, and HFE principles, guidelines, and standards; develop and perform HFE analyses; identify and participate in the resolution of identified HFE issues and non-compliances. This role requires a bachelor's degree in HFE, Engineering Psychology, or related science with four years of cumulative experience related to the HFE aspects of HSIs (design, development, and T&E), particularly modern digital process control HSIs, and four years of cumulative experience related to the HFE aspects of workplace design.
3. HFE Operations/Maintenance: Provide knowledge of operations and maintenance activities, including task characteristics, HSI characteristics, environmental characteristics, and technical requirements related to operational activities, and apply those insight in support of activities such as development of HSIs, procedures, and training programs. Participate in the development of scenarios for Human Reliability Analysis (HRA) evaluations, task analyses, HSI T&E, validation, and other evaluations. This role requires a bachelor's degree in a technical field; experience as a senior authorized reactor operator, or as a qualified maintenance technician; five or more years of plant experience, preferably in Boiling Water Reactors exposure to plant procedure development, personnel training, and operational nuclear plant programs; and two or more years of experience in one or more areas of HFE analysis, design, T&E, and HFE V&V.

The responsibilities of the HFE team are to establish and perform the activities as defined in this PSAR chapter throughout the design lifecycle to ensure that the facilities, systems, equipment, and tools are designed to be compatible with the capabilities, limitations, and needs of the human. The specific duties of the HFE team are to guide, perform, and support the analysis and design activities, ensuring the execution and documentation of all steps in the activities are performed in accordance with the established program and procedures.

The HFE team is responsible for:

1. Development of HFE plans and procedures including management of any identified HFE issues
2. Oversight, participation in, and review of HFE design, safety analyses, development, T&E activities, including identification of in-process HFE issues
3. Recommendations for, and support to implementation of design-based resolutions for issues identified during the implementation of the HFE requirements and analysis results
4. Verification of correct and robust implementation of HFE requirements, analysis results and issue resolution into the design
5. Assurance that HFE activities comply with HFE plans and procedures
6. Managing documentation of HFE activities and issues management
7. Plan and implement HSI design configuration control during design implementation

To ensure suitably qualified and experienced persons are performing the work, HFE Program activity assignments are allocated by the Technical Lead based on the team member's role and their specific experience. For example, not all the team HF Engineers/HFE Specialists are experienced in HFE safety analysis, identifying, and evaluating important human actions. Only those team members with adequate experience and training (if applicable) in a particular HFE activity will be assigned to those activities. To ensure there are no singleton specialisms within the team and associated vulnerability to knowledge loss, more than one team member will be required to be deemed competent for each activity. Team members that do not meet the full qualification of an HFE team role, or who are not deemed suitably qualified for a specific activity, will receive mentoring and technical oversight to support developing the skills required for the role or work assignment.

#### **18.1.3.1 Cross-Discipline Support and Integration**

Due to the cross-functional nature of a completely integrated HFE design process, HFE activities interface with many other disciplines. In addition, the other disciplines act as extended parts of the HFE team for some aspects of the HFE Program implementation. The integration of related groups with HFE is formally addressed through an integrated detailed schedule, as well as through the HFE technical project management role of the HFE Technical Lead.

Specifically, work activities that require integration of HFE and other disciplines are entered into the resource-loaded schedule by the HFE Technical Lead with all required resources, including those from other teams. Pre-job briefs are held for each group of activities, or workplan, to clearly define the relevant resource roles and responsibilities for the completion of the work. Further detail on GEH and the BWRX-300 project design processes, including inter-discipline communications and issues resolution, are described in Chapter 17, Sections 17.2 and 17.3.

The descriptions of the following disciplines and groups and their contributions to HFE are representative based on best-practice HFE integration principles. The actual engineering design team disciplines may vary, but the scopes described are covered.

1. Mechanical Engineering and Electrical Engineering
  - a. Provide knowledge of the purpose, operating characteristics, and technical specifications of major plant systems
  - b. Provide input to HFE analyses, especially function and task analyses

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
  - d. Participate in developing scenarios for use in TA, validation, and other analyses
2. I&C Engineering
- a. Provide detailed knowledge of the HSI physical design, including control and display hardware selection, design specification, functionality, and installation
  - b. Support HFE design of information display design, content, and functionality, particularly connection to the underlying I&C platform
  - c. Participate in designing, developing, testing, and evaluating the HSIs
  - d. Provide knowledge of data processing associated with displays and controls
  - e. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
  - f. Participate in designing and selecting HSI components, such as controls and displays
  - g. Participate in developing scenarios for HRA, validation, and other analyses involving failures of the HSI data processing systems
3. Civil/Structural Engineering
- a. Provide knowledge of the overall structure of the plant, including performance requirements, design constraints, and design characteristics of the following:
    - Containment structures (i.e., Steel-Plate Composite Containment Vessel)
    - Control Rooms (main and secondary)
    - Local control
  - b. Provide knowledge of the configuration of plant components
  - c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
  - d. Provide input to plant analyses, especially function analysis, TA, and development of scenarios for TA and validation
4. Plant Integration Engineering
- a. Prepare the Deterministic Safety Analysis (DSA) establishing the SSC and human actions that are credited for successful event mitigation
  - b. Provide knowledge of maintenance, inspection, and surveillance activities based on previous plant design and consideration of evolving new plant design, including:
    - i. Development of maintenance and outage strategy and plan documents
    - ii. Expected tasks
    - iii. Relevant SSC and HSIs
    - iv. Task performance requirements
    - v. Workspace environment characteristics
    - vi. Technical information related to the conduct of these activities

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
  - d. Support Plant Architectural and HFE design, development, and evaluation of the control facilities and other HSIs throughout the plant to provide reasonable assurance that each can be inspected and maintained to the specified reliability
  - e. Provide input regarding maintainability and inspectability during the development of procedures and training
  - f. Participate in the development of scenarios for HSI evaluations including task analyses, HSI design tests and evaluations, and validation
5. Risk and Reliability Engineering
- a. Perform Probabilistic Safety Assessment (PSA) and HRA to quantify the human contribution to risk and inform HFE analyses
  - b. Provide knowledge of plant component and system reliability and availability and assessment methodologies to the HSI development activities
  - c. Participate in the development of scenarios for HSI evaluations, especially validation
  - d. Provide input to the design of HSIs to provide reasonable assurance it meets reliability goals during operation and maintenance and maintains the specified availability
6. Simulation Assisted Engineering
- a. Develop the simulators for HFE T&E and V&V activities

#### **18.1.4 Process and Procedures**

##### **18.1.4.1 Coordination and Documentation of Activities**

The HFE Program is planned and conducted in accordance and alignment with overarching design and quality program processes and procedures, within accredited quality management systems as described in Chapter 17. The work is performed in an integrated manner with HFE as an equal design discipline, whose cross-cutting requirements and support are embedded within all other design disciplines, with activities, inputs, outputs, and dependencies coordinated through a detailed schedule and associated schedule management processes. The schedule includes activities and deliverables for all disciplines and orders them with logical connections to ensure they are completed in the required sequence.

To help ensure cross-discipline communication and coordination, activities include scheduled periodic formal design reviews conducted by representatives of each discipline. Additionally, deliverables are completed in accordance with a deliverable standard, which specifies the required content from all related disciplines, dictates the format for consistency and quality and specifies the required discipline reviewers for each document. This includes other disciplines reviewing and incorporating outputs from HFE documentation and the HFE team reviewing and incorporating outputs from other disciplines, as appropriate and according to the plan.

The documentation for the HFE Program uses a standard design process that includes documenting internal design records to capture inputs and outputs, as well as providing the basis for formal deliverables. The information in the design records is incorporated into the design by HFE and other disciplines as appropriate and in accordance with the BWRX-300 project work breakdown structure. A full description of the management and integration of HFE activities within the project is described in the HFEPP.

In this way, using the project processes and HFE coordination measures described in the HFEPP, the design activities related to HFE are conducted and documented such that design basis, input maturity and rationale for design and analysis scope is provided for HFE design decisions and analysis results. HFE requirements and recommendations are addressed by the requirements management process (Subsection 18.1.4.3), and either incorporated into the design directly or via alternate solutions agreed as acceptable by HFE, or if not implemented, tracked as an HFE issue as described in Subsection 18.1.5.

#### **18.1.4.2 Risk-Based Graded Approach**

A graded (or proportionate) approach to HFE is applied to the conduct of activities within the HFE Program, to provide the appropriate focus for analysis and design. The graded approach provides basic HFE attention to human interactions within the system and provides emphasis and more detailed, rigorous HFE effort on aspects of the plant design related to HSIs used to perform human actions important to safety, or tasks that are novel, complex, or inherently hazardous. The approach uses a risk-based grading system to grade each of the tasks or human actions identified throughout the plant based on four key risk categories:

- Nuclear Safety
- Personnel Safety
- Asset Protection
- Generation Capability

Although Asset Protection and Generation Capability are typically discounted as relating to safety, the HFE Program recognizes that equipment damage creates requirements for forced outages and corrective maintenance, as well as impacting production goals, all of which has an indirect impact on personnel and nuclear safety. Consideration of aspects of equipment protection, reliability, and production risks, also ensures that tasks outside of reactor operations have a decreased likelihood of being classified as Low-Risk Level. Loss of power generation and production shortfalls equate to loss of income for the plant which is recognized to have a direct effect on plant condition, organizational health, and ultimately, nuclear safety culture.

The overall risk level for the human action is determined by the highest risk level assigned to each of the four categories. The base risk level is then used to assign a minimum HFE Application Level. The minimum HFE Application Level dictates the minimum degree of application when considering each HFE technical element. Further detail is provided in Subsection 18.2.3.

In addition to the formal task grading method for determining proportionate effort during design, the scope and level of effort of HFE activities is also proportionate to project lifecycle risk and change management considerations. This is done by applying greater scope, focus and degree of support on HFE activities that occur earlier within the design lifecycle, when changes are more effectively and easily managed. For example, while not all HSIs receive a full HFE V&V, all HSIs receive some degree of HFE support during the design phase.

#### **18.1.4.3 Requirements Management**

HFE requirements management is performed in accordance with a requirements management process that is standardized and controlled across the entire plant design, as described in Chapter 17, Subsection 17.3.1. Requirements management and traceability is developed and maintained to:

- Ensure HFE requirements relevant to each scope of work are clearly identified, allocated, communicated, and understood by all relevant project personnel

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- Outline how the requirements are met
- Reference evidence that demonstrates compliance has been achieved

HFE requirements are categorized and dispositioned as follows:

- A. Process Requirements – Requirements related to how the HFE Program is conducted and the interfaces among HFE and other disciplines. HFE Process Requirements are allocated to the HFEPP and are fulfilled via the construction and implementation of the architecture and processes defined within this document. As such, all HFE Process Requirements are allocated to and end with the HFEPP.
- B. Product Requirements – Requirements for the design of, or provision for plant workspace and environmental attributes, SSC, and HSIs. Product requirements are either derived from HFE design standards, codes, and guidance, or generated from HFE analyses as required to support successful task performance in the specific context of plant conditions. These requirements are implemented through design requirements specifications or design records communicated to and implemented by the relevant design teams. Requirement traceability and HFE support level is proportionately applied based on the assessed risk-based grading (Subsection 18.4.2).

Where HFE inputs are not in agreement with one another or where they conflict with other discipline design requirements, precedence is given as follows:

- Laws and Regulations
- Regulatory Requirements and Guides
- Requirements related to supporting, maintaining, or recovering the plant in a safe state
- Nuclear Standards
- Nuclear Industry Guidance Documents
- Non-Nuclear Codes, Standards, and Guides

Conflicts between HFE and other design requirements are resolved with the HFE team using the HFE issue resolution process described in Subsection 18.1.5. Decisions regarding trade-offs and design optimizations are conducted within the integrated design process in accordance with standard GEH design procedures.

#### **18.1.5 Issue Tracking**

Included in the HFE Program is the establishment and maintenance of an on-going HFE Issues Tracking System (HFEITS) for documenting HF issues that may be identified throughout the full scope of HFE activities, and the actions taken to resolve those issues.

The HFEITS is used to capture issues related to design and implementation HFE activities and specific Human Engineering Discrepancies (HEDs) identified through HFE V&V. The tracking of issues include:

1. Evaluation of each issue/HED to determine significance and whether it warrants correction when evaluated in the context of the integrated plant design
2. Identification of appropriate solutions to address issues/HEDs, including, as appropriate, changes to HSI design, procedures, staffing/qualifications, or training
3. Verification that the solutions implemented to address the issue/HED resolve the problem without generating additional issues/HEDs



NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

4. Documented traceability of the issue/HED resolution process and identification of residual risks associated with it, if necessary

The HFEPP details the HFEITS development and management including:

1. Responsibilities for HFE team members in identifying HFE issues and HEDs
2. The process and criteria for including HFE issues and HEDs within the HFEITS, as opposed to resolving non-compliant design through integrated design teamwork in normal workflow
3. The process for evaluating the priority and adequate resolution of the issue/HED
4. The means for confirming acceptable resolution of the issue/HED, based on the nature of the issue, its priority, and the plant lifecycle stage where the resolution occurs

#### **18.1.6 References**

- 18.1-1 NEDC-33982P, "BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan," GE-Hitachi Nuclear Energy Americas, LLC.
- 18.1-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 18.1-3 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.1-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.1-5 CSA N290.12-14, Human factors in design for nuclear power plants, Canadian Standard Association.

## **18.2 Human Factors Engineering Analysis**

In this section, six of the HFE Program technical elements are described in depth. The remaining elements are described in other sections. These six elements described in this section include:

- Review of Operating Experience (OE)
- Function Requirements Analysis
- AOF
- TA
- Staffing
- Treatment of Important Human Actions

These are addressed in Subsections 18.2.1 through 18.2.6.

Integrated HFE involvement in, and support of the plant safety analyses also informs these elements. As described in Subsection 18.2.6, the HFE safety analyses activities identify, characterize, and substantiate the human actions that are performed to maintain the plant within or bring it back to a safe state, as described in Chapter 15, Safety Analysis.

### **18.2.1 Review of Operating Experience**

The HFE Program includes the early review of OE to identify applicable HFE issues related to process or personnel safety that can be resolved through design improvements. The issues and lessons learned from the OER provide a basis for improving the plant design in a timely way (i.e., at the beginning of the design process). In addition to the early HFE review of existing OE, the project as a whole has a formal OE identification and management process, which includes HFE team participation in both identifying any OE and implementing HFE-allocated OE items in the HFE Program and design.

#### **18.2.1.1 Objectives and Scope**

The objective of the OER is to obtain information and lessons learned from experience to support design of BWRX-300 SSC. OE related to the following areas are considered in the development of the plant design:

- Predecessor plant(s) and systems
- Experience in industries with applicable SSC
- Applicable Industry HSI design experience
- Risk-important human actions
- Specifically identified applicable industry issues
- Issues identified by predecessor or similar plant personnel
- Specifically identified positive features that support task performance

#### **18.2.1.2 Methodology**

The OER process includes the following:

- Identification of applicable OE sources, leveraging work previously performed for predecessor plant designs

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- Specific methods for gathering sources and performing reviews of sources since the predecessor plants and systems OER
- Establishing a systematic framework and performing systematic searches of OE sources
- Obtaining and incorporating personnel feedback from predecessor or similar types of reactors
- Conducting reviews of human actions from predecessor designs that are similar to human actions included in the plant safety analyses
- Analyzing and consolidating raw OE data into OE Item Summaries on the OER Capture Sheet
- Allocation of each OE item to the HFE activity or document in which the OE item is dispositioned
- Recipient OE item review and allocation acceptance
- Documentation of the findings within an OER report

Existing and new OE is reviewed by HFE, and relevant, applicable problems, issues, and positive insights are identified and addressed throughout the design process. The OER information is made available to design engineers to support development of design features that are expected to reduce human error. Likewise, positive features of previous designs are communicated so that they can be retained.

#### **18.2.1.3 Results**

The results of the OER are summarized in an OER report. The report provides the OER process description along with the review methods that were used. The results include:

1. Sources of OER information
2. Summaries of OER issues and improvements
3. List of issues from the OER requiring special attention in the design process based on the grading process
4. Information gathered from personnel interviews conducted at predecessor plants

Implementation of the OER results into the design is managed and tracked through the assignment of and reference to a unique OE identification number. Communication and allocation of the results to the appropriate design team and design requirements document is managed by the HFE team.

The HFEPP provides additional details of the OER activity.

### **18.2.2 Functional Requirements Analysis**

#### **18.2.2.1 Objectives and Scope**

FRA is performed to define the necessary functions that enable the achievement of the plant safety and commercial goals. These principal design requirements are necessary to meet plant goals and objectives in all normal and postulated accident conditions. They include meeting regulatory and customer requirements that are documented in plant and product level design requirements specifications.

### **18.2.2.2 Methodology**

FRA is conducted as an integral part of the overall engineering design process specified by standard design process documents, and in particular part of the BWRX-300 requirements management process. The requirements management process as described in Chapter 17, Subsection 17.3.1 consists of the following activities, which apply equally to all types of requirements including the functional ones: elicitation, analysis, documentation, allocation (to specific system(s)), tracing, and requirements V&V. This process is a multi-discipline activity jointly undertaken by all design teams, including the HFE team. Note that, because BWRX-300 is an evolutionary plant with some of the same or similar plant- and system-level safety and performance goals, FRA elicitation is not required to the extent it would be for a completely new design. Eliciting functional requirements from existing design is done through importing functions that exist in current plant designs and are expected to apply to BWRX-300.

### **18.2.2.3 Results**

The result of the multi-disciplinary FRA activities is the definition of the full set of functions that support achievement of the plant goals and can be traced to the principal design requirements of the BWRX-300. Plant- and system-level requirements documents list the functional requirements associated with each system. Through the requirements management process described in Chapter 17, Subsection 17.3.1 the project has also captured system functional performance requirements which includes the full list of plant functions, with reference and traceability back to the source documents where the requirements were elicited.

These functions from the FRA are captured in a report that also includes the results from the AOF (see next section) and assigning HFE Application Levels to tasks (Subsection 18.1.4.2). The results are input into the AOF process as described in the next section. The output from the FRA and AOF process also contains the full list of functions with characterizations of relevance to HFE, particularly those required to perform AOF and feed to the TA and HSI design activities.

### **18.2.3 Allocation of Function**

#### **18.2.3.1 Objective and Scope**

AOF establishes a plant control scheme that enhances plant safety and reliability by taking advantage of human and system strengths and avoiding human and system limitations. The overall allocation can also enhance plant performance and safety by specifying overlapping and redundant responsibilities to the human and system.

The AOF strives to provide personnel with groups of logical, coherent, and meaningful tasks within their capabilities, and ensures a design that maintains human vigilance and situational awareness for any functions allocated to the system. The goal of the AOF is to provide acceptable workload levels per job role that minimize periods of human underload and overload to the extent possible. This is done through review of the initial allocation as a whole and using expert judgement to determine if the assigned functions per job role are suitable and sufficient. Further analysis of workload and requirements for situational awareness are then undertaken through downstream activities such as TA, HFE T&E and HFE V&V.

The AOF also allows the risk-based task grading, which determines the HFE proportionate, graded approach to activities, as described in Subsection 18.1.4.2.

#### **18.2.3.2 Methodology**

The AOF process for BWRX-300 is based on the relevant best-practice methodology presented in IAEA-TECDOC-668, "The role of automation and humans in nuclear power plants" (Reference

18.2-1). The methodology identifies functions that should not be assigned to humans due to criteria such as:

- Physical demands (forces, posture)
- Cognitive demands (multitasking, stress, situational awareness, and vigilance)
- Combination of physical and cognitive demands (accuracy, response time)
- Environmental conditions (temperature, radiation)

The AOF process also uses criteria from USNRC NUREG/CR-2623, "The Allocation of Functions in Man-Machine Systems: A Perspective and Literature Review," (Reference 18.2-2) that limit or preclude human participation in a function or, conversely, that make human participation mandatory. These combined criteria form the top-level, overriding criteria in the AOF process.

The FRA process provides the list of functions input to the AOF, as described in Subsection 18.2.2. The AOF process is composed of two stages: the first stage establishes an initial (hypothesized) allocation; the second stage evaluates the hypothesized AOF to determine its adequacy and validate that it is optimized within the larger integrated task performance and work environment.

The initial AOF is determined following a formal decision flow with key allocation criteria informing each decision point. The decisions are based on expert judgement formed by a panel that includes an HF Engineer/HFE Specialist, an HFE Operations/Maintenance representative, a Plant Integration Engineering representative, and a System Engineer for the respective system. The expert panel accounts for the mandatory criteria when hypothesizing the initial allocation. The panel also makes use of OE to determine how functions were allocated in previous or similar applications and evaluate how they have performed.

The second stage of the AOF process is the AOF evaluation. This stage is performed later in the design and HFE T&E process, following completion of system level and integrated TA. The AOF evaluation is a structured examination of function and task groupings that is used to assess allocations in a collective manner within an integrated work environment, instead of on a single function basis, where overload issues are less likely to be revealed. Functions and tasks allocated to humans are considered in combination using scenario development. The scenarios are then evaluated to determine acceptability based on expected concurrent task performance, workload (physical and cognitive), vigilance, and situation awareness.

As part of the AOF activity, functions are decomposed to tasks, and multiple tasks may be necessary to support each function. For example, to fulfil a core protection function, a task to perform a system readiness surveillance test supports the eventual task of safety system initiation. The function tasks require a "task allocation" that uses the same criteria applied at the overall AOF level. The safety analyses also provide input to the AOF, specifying when human actions are required to backup automatic (i.e., system) actions.

In addition to allocation, within the AOF activity, task grading is completed. As described in Subsection 18.1.4.2, a graded approach to HFE is applied to the BWRX-300 project. The human actions resulting from the AOF process described above are graded based on four key risk categories:

1. Nuclear Safety
2. Personnel Safety
3. Asset Protection
4. Generation Capability

The characteristics of the human actions are assessed against criteria in each category that results in an initial High, Medium, or Low numerical risk rating. This gives the overall minimum HFE Application Level for activities within the HFE Program for each human action. The scope and level of detail for each HFE technical element is defined for each of these HFE Application Levels.

When the risk is assessed numerically using the initial ranking criteria, the result is a minimum HFE Application Level. The HF Engineer/HFE Specialist reviews the rating and the associated HFE scope and level of effort for each technical element and takes into consideration other HFE risk factors to determine if the HFE Application Level needs to be increased for that specific technical element. The additional risk factors include:

- Complexity of the action
- Anticipated complexity and constraints of HSI
- Complexity of the system
- Frequency of the task
- Physical environment
- Cognitive environment
- Novelty of the action, system, or HSI technology
- Time sensitivity of the action

The review of an individual HFE technical element for each human action may increase an HFE Application Level but not reduce it. For example, a particular human action may have a low minimum application level based on the broad key risk criteria, but because it is a complex or novel task, it is raised to a higher application level for the TA activity.

#### **18.2.3.3 Results**

Output from the first stage of AOF is the initial AOF to human, system, or shared (both human and system). For system or shared allocations, it may be necessary to establish backup actions when redundant functions with like allocations is not possible or reasonable. For these cases, the shared and backup allocation categories are used.

The initial allocation for each system AOF is documented in a workbook which is formally managed as an internal design basis record. The workbook is communicated to all relevant stakeholders (Mechanical and I&C engineers, as well as HFE analysts) to be used as the basis for the design and TA. The initial allocation is revised and refined as necessary as the plant design and safety analyses progresses.

The results from the AOF evaluation are a final refinement of the initial allocation, including design and safety analysis modifications where necessary to support changes to allocation outcome. The final optimal AOF is used for Integrated System Validation (ISV) testing to confirm that performance, workload, and situation awareness are suitable.

The results of the AOF development and refinement activities, and specification of the final AOF are provided in a summary report.

#### **18.2.4 Task Analysis**

TA is the identification of task requirements to accomplish the functions and tasks that have been allocated in whole, or in part, to humans. These are designated in the AOF results as Human, Shared, or System with Human Backup.

TA assigns tasks to the job positions specified by the staffing process. TA determines the steps needed to accomplish human actions and documents the task details and required task support (HSI controls, indications, and alarms). The TA process also assesses the graded HFE Application Level to determine if a change to is warranted based on task characteristics.

#### **18.2.4.1 Objectives and Scope**

The TA plan establishes:

1. Methods for conduct of the TA consistent with accepted HFE practices and principles
2. Scope of the TA including actions performed at the MCR, SCR, and at other control facilities, including those required to support response to accidents and emergencies
3. Range of plant operating conditions, including start-up, normal and abnormal operations, transients, refuelling, lower power, and shutdown conditions, and emergency or accident conditions
4. HSI operations during periods of maintenance, testing, and inspection of plant SSC
5. Links among task descriptions and safety importance, function achievement, human error potential, and impact of task failure
6. Descriptions of the personnel activities required for successful completion of tasks
7. Requirements for alarms, displays, data processing, and control

#### **18.2.4.2 Methodology**

The task inputs provided by the AOF and Task Grading process form the starting point for TA. These tasks are divided into levels of effort as defined through the task grading portion of the AOF process, as described in Subsection 18.2.3. All tasks regardless of HFE Application Level receive a TA. However, the level of detail within the TA varies based on HFE Application Level. For example, those at the lowest risk level may be performed by the responsible System Engineer and reviewed by HFE for acceptability.

There are two levels of TA: Basic TA and Detailed TA. Human actions ranked at the lowest HFE Application Level undergo a Basic TA; human actions at the medium and highest HFE Application Levels undergo a Detailed TA. The Detailed TA also includes preliminary workload analysis and assessment of requirements for situational awareness. In addition to the Detailed TA, the highest HFE Application Level requires additional link Analysis and timeline Analysis to be performed to evaluate and inform the layout of HSIs to optimize task performance.

Basic TA consists of:

- Task Selection
- Task Step Sequence Narrative, including:
  - Descriptive narrative of the task
  - Cue that determines the need for the task
  - Action to be taken
  - Prerequisites for the task
  - Time available versus time required to complete

Detailed TA consists of:

- Task Selection

- Task Step Sequence Narrative, as per Basic TA
- Task-Level Support, Job Design, Workload, and Workplace Definition including:
  - Information needed
  - Controls needed
  - Alarms needed
  - Personnel involved
  - Communication needs
  - Location and access considerations
  - Workspace needed
  - Job aids, tools, or equipment needs
  - Environmental considerations and potential hazards
  - Special clothing or personal protective equipment needs
  - Time available versus time required to complete (if needed on a step basis)

The Detailed TA is coordinated with the qualitative human error analysis, which is informed by and used to substantiate the quantitative HRA, as described in Subsection 18.2.6. The TA also identifies critical task steps where incorrect or incomplete actions might lead to undesired or unsafe consequences.

The TA activity also includes Integrated TA. Integrated TA is conducted for those operations that require interaction with multiple systems and a coordinated response that may involve multiple plant personnel. The Integrated TA takes the system-level TAs and the results from the human error analyses (described in Subsection 18.2.6) and performs higher-level whole sequence analyses, including integrated workload analysis and timeline analysis. The Integrated TAs, at a minimum, include each event sequence that contains an HFE Application Level 1 human action and each unique scenario described in the plant safety analyses.

#### **18.2.4.3 Results**

The BWRX-300 TA procedure defines set templates for capturing the TA, which are combined with the AOF results workbook, documenting the full set of analysis activities. The activities are iterative and progressive, and the workbook method allows timely and effective update of the TA and distribution of the results. The TA workbook is functionally divided into subsets of information needed to process the output activities from TA, including input to:

- Staffing and qualifications
- HSI development, including I&C Data Connection Tables (DCTs)
- Training development
- Procedure development

The results from the additional link, timeline and preliminary workload analyses in the Detailed TAs are also used to further inform the HSI design requirements, confirm or identify issues with the AOF, and provide a baseline for HSI T&E activities. The Integrated TA results provide an input into group-use and aggregate HSI designs, control facility and plant location and workspace design considerations, development of plant-level procedures, and the scenarios selected for ISV.



The results of all the TA activities, when they are complete, are summarized in a TA summary report.

### **18.2.5 Staffing**

The required and expected number of personnel available to achieve plant functions and goals is an important consideration throughout the design process and in HFE analyses. BWRX-300 staffing assumptions and analysis results are used to frame the future plant operating organization during TA, HRA, and HSI design.

Features of the BWRX-300, such as passive safety systems, increased automation, and simplified HSIs, information systems and content, decrease the assumed initial staffing requirements relative to previous Boiling Water Reactors. For example, TA done as part of the HFE safety analysis work, described in Subsection 18.2.6, may show that the extended time for safety actions may reduce the number of personnel needed for local actions. Safety analyses and identification of human actions may show that some actions that were important in previous boiling water reactor designs have been eliminated in the plant design.

#### **18.2.5.1 Objectives and Scope**

Staffing analysis is conducted to determine the minimum staff complement. The minimum staff complement is defined as the minimum number of workers with specific qualifications who are available to the site at all times. The minimum staff must be able to operate and maintain the plant within its defined safe operating envelope and to successfully respond to all postulated events in the safety analyses, in any plant state.

For staffing assumptions where minimum staff can be varied for different operational states, the most resource-intensive events for each plant mode are analyzed. The Staffing Analysis technical element also determines the maximum staffing in collaborative areas such as control rooms, workshops, and air locks. This informs the needed space, facilities, and other support features.

#### **18.2.5.2 Methodology**

The Staffing Analysis for BWRX-300 is conducted in accordance with a plan that was developed to meet the expectations and requirements of CNSC REGDOC-2.2.5, "Minimum Staff Complement" (Reference 18.2-3). The process takes place in three major steps: Expert Panel Staffing Assessment; Staffing Analysis in TA and HSI design; and Staffing Analysis in the HFE V&V Process.

The process starts with an assumed initial staff complement taken from predecessor and similar plants, and representative OE from the operating fleet. Using this information, the initial staffing is optimized, considering the modern design features and new systems of the BWRX-300.

The optimized initial staffing level is subject to an Expert Panel Staffing Assessment that evaluates the minimum staffing to cope with selected credible events. The goal is to perform early and iterative assessments as the design progresses such that the risk of less than adequate staffing is reduced in part with each evaluation. The evaluations are performed as a desktop (talk-through) exercise led by the expert panel. The expert panel is made up of personnel from the HFE team, supported by personnel from Plant Integration Engineering and Risk and Reliability Engineering.

The next steps take place in conjunction with the TA and HSI design process. During TA, task steps are defined, and personnel assignments are made. The TA forms the basis for job design and qualifications for each role. With the TA, timeline analysis is conducted for the most resource-intensive credible events. These events are selected by the expert panel, and the expert panel

also performs a review of the timeline analysis and input TA data to evaluate whether the minimum staff complement is adequate.

After this, scenarios for a wider range of events are created and analyzed in the T&E phase of the TA and HSI design process. This allows further evaluation of the minimum staff complement against the most challenging credible events. HSI tests evaluate, using platforms and mock-ups that replicate the interface design, the timing of activities, workload, and other factors such as situation awareness that can lead to changes in the minimum staff complement and job design.

In the final analysis stage, the minimum staff complement is evaluated through the HFE V&V activities. This occurs during early validation in accordance with the multiphase validation approach (Section 18.4), culminating with the ISV. ISV demonstrates the adequacy of the final staffing levels that resulted from the analysis.

#### **18.2.5.3 Results**

The staffing analysis results are recorded in a series of reports capturing each Expert Panel Assessment, and a further report capturing the Expert Panel Review of the related staffing analysis activities. In addition to these reports, outputs and reports that are created for the TA and HSI design process and HFE V&V in accordance with those associated technical element descriptions, include results related to or impacting the staffing analysis results.

The confirmed job role and complement determination is also used as an input to training and qualification program development, where base qualifications are established, and the training program is designed (Subsection 18.3.8).

#### **18.2.6 Treatment of Important Human Actions**

Consideration and integration of HFE within the safety analyses, and consideration of the results and assumptions of the safety analyses within the other HFE activities both comprise the technical element of Treatment of Important Human Actions. The set of activities supporting this HFE technical element was developed based on requirements for HFE expertise in safety analyses per IAEA SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants" (Reference 18.2-4), CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.2-5), CSA N290.12-14, "Human factors in design for nuclear power plants" (Reference 18.2-6), and international best practice.

##### **18.2.6.1 Objectives and Scope**

HFE safety analysis activities, including review of safety analyses outcomes, provides assurance that the full set of human actions that are important to safety are explicitly identified, characterized, and substantiated as achievable within the task performance requirements.

The important human actions are determined using both deterministic and probabilistic means and include identification of the human actions that might cause or contribute to the cause of postulated initiating events. Inclusion in the DSA, PSA, or Severe Accident Analysis (SAA), or other identified contribution to risk determines the risk level in the Nuclear Safety category for determining the initial HFE Application Level to apply to the HFE activities (Subsection 18.1.4.2). Comprehensive, systematic identification and substantiation of human actions claimed within the safety analyses, coupled with the risk-based graded approach described in Subsection 18.1.4.2, ensures that HSIs and tasks associated with important human actions are analyzed and designed with a full detailed and robust HFE effort.

#### **18.2.6.2 Methodology**

The safety functions that are performed to maintain the plant within or return it to a safe operating envelope are identified through various means using DSA, PSA, and SAA, as described in Chapter 15, Safety Analysis.

HFE safety analysis activities use the various safety analyses, available system design information, particularly that related to HSIs, and any available and applicable TA results (Subsection 18.2.4) as inputs.

The activities that form part of this technical element include:

1. Perform a Human Operation Hazard Evaluation
2. Review the DSA for explicit and implied human actions, e.g., human actions related to maintenance that ensure safety-class SSC availability, and ensure claimed human actions are achievable through qualitative human error analysis
3. Review the PSA and HRA for:
  - a. Ensuring event sequences introduce a Human Performance Limiting Value
  - b. Perform qualitative human error analysis to substantiate or refine the human error probabilities used for all human actions claimed in the HRA
  - c. Provide HFE qualitative basis and substantiation for any dependency assumptions and analysis
  - d. Provide HFE qualitative basis and substantiation for any timeline assumptions and analysis
4. Review the SAA, including Level 2 and Level 3 PSA, to identify explicit and implied human actions, determine task achievability in required timescales, and capture input for procedure development (Subsection 18.3.7) and emergency planning and response (as described in Chapter 19)
5. Compile a database of all important human actions claimed in the safety analyses, including the source of the claim, their key characteristics, related assumptions and any associated HSIs

For each of the above activities, the task performance criteria are defined to enable the HFE analysis to determine the acceptable achievability of each identified human action. Task performance requirements depend on the plant conditions the task is performed in (normal versus abnormal versus emergency or accident conditions), and for events, are defined by the related safety analysis the HFE analysis is underpinning. For example, for precursor human actions, performance requirements are usually "performed correctly" or "performed in accordance with maintenance schedule timelines". For post-initiating event human actions, the requirements are defined by the event conditions. Tasks must be achievable, completed successfully and, where dictated by the related analysis, must be completed within the required time based on the event timeline.

#### **18.2.6.3 Results**

The outputs from the activities that form this technical element includes:

1. A Human Operation Hazard Evaluation report that documents the methods used and the evaluation results
2. Design records capturing results of safety analyses reviews

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

3. Design records capturing HFE qualitative human error analysis and other substantiation of claimed human actions in the various safety analyses
4. A Human Action Claims database containing all important human actions and critical information for each, allowing for adequate and robust consideration in the design
5. A summary report, which summarizes the methods, results, and issues from each of the above outputs

The results are communicated directly to the appropriate design, safety analysis and other HFE team members performing related technical elements, to ensure timely consideration of the full set of human actions important to safety in their respective activities. The outputs from each activity and the Human Action Claims database capture the source of the identified human actions within the safety analysis to allow full traceability from the origin through HFE safety analysis and forward to any design requirements or safety analysis modifications.

In addition to specific HFE safety analysis activities described in this section, the full complement of safety analysis outcomes, including the PSA and HRA, informs and acts as input to the other HFE technical elements, as applicable.

#### **18.2.7 References**

- 18.2-1 IAEA-TECDOC-668, "The role of automation and humans in nuclear power plants," International Atomic Energy Association.
- 18.2-2 USNRC NUREG/CR-2623, "The Allocation of Functions in Man-Machine Systems: A Perspective and Literature Review," U.S. Nuclear Regulatory Commission.
- 18.2-3 CNSC Regulatory Document REGDOC-2.2.5, "Minimum Staff Complement."
- 18.2-4 IAEA Safety Standards Series No. SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants," International Atomic Energy Association.
- 18.2-5 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.2-6 CSA N290.12, "Human factors in design for nuclear power plants," CSA Group.

### 18.3 Design of the Human-System Interface

This section describes the process by which HSI designs are established and evaluated. The HSI design process for BWRX-300 is governed by a process methodology report that outlines the required design inputs, design procedure to be followed, design outputs, and the process for conducting HFE T&E during design development. The HSI design process sits within and is fully integrated into the overall plant design process specified through standard engineering design process and procedure documents, as well as the relevant project-specific design process plans. General design principles and processes are described in Chapter 3, particularly Subsection 3.1.7.

#### 18.3.1 Design Goals and Design Bases

The primary goal of the HSI design process is to facilitate safe, efficient, and reliable user task performance during plant normal operational states, abnormal events, and accident conditions. To achieve this goal, HSIs throughout the plant are designed and implemented consistent with HFE core principles and user-centred design practices. The following specific design bases are adopted for the plant:

1. HSI design promotes efficient and reliable operation through application of automated operation capabilities.
2. HSI design uses only proven technology.
3. The workstation and HSI layouts reflect I&C separation restrictions.
4. HSI design is highly reliable and provides functional redundancy such that sufficient displays and controls are available in the MCR and SCR and remote locations to conduct an orderly reactor shutdown and to cooldown the reactor to safe shutdown conditions, even during design basis equipment failures.
5. The principal functions of the Safety Parameter Display System (SPDS) as required by CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.3-1) are integrated into the HSI design.

*NOTE: Historically, the SPDS has provided an overview display of important plant parameters during transients and accidents on a separate panel with separate safety categorization due to the safety analysis outcomes for full-scale plants. The SPDS terminology is used in this PSAR for consistency with regulatory documents. However, the BWRX-300 SPDS functionality does not require a different safety classification and is an integral part of, and included with the SC3 displays (see Subsection 18.3.5). There is no separate panel or system.*

6. Accepted HFE principles and methods are used for ensuring HFE is integrated into the design, in alignment with international best practice and meeting the requirements of CNSC REGDOC-2.5.1, (Reference 18.3-2).
7. HFE design requirements are based on international standards and applicable CNSC regulatory requirements, as outlined in the HFEPP.
8. The design basis for accident and emergency control and monitoring facilities meets international standards as well as CNSC REGDOC-2.5.2 (Reference 18.3-1), and CNSC REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response" (Reference 18.3-3).

Task performance criteria developed through the HFE Program analysis activities, in conjunction with those described in Chapter 3, Section 3.1 and Chapter 15, Subsections 15.3.2 and 15.3.3

are used to govern and direct HSI design specifications. These detailed task performance criteria, along with requirements specified in HFE standards and codes, encompass the set of necessary and sufficient design requirements that maintains the implemented plant SSC and HSIs in compliance with accepted HFE principles.

### **18.3.2 Human-System Interface: Design Inputs**

The inputs to the HSI design are derived from several sources. These include specific information relating to the performance of tasks, as well as design requirements and guidance specific to the plant design and a defined full set of plant user characteristics, but not related to task performance. In addition to these documented input sources, the integrated set of HFE Program activities includes HF Engineer/HFE Specialist support to designers for instances where the correct application of the set requirements is not clear or where design conflicts exist, and suitable alternative design solutions are required. Finally, HSI design updates are made based on results from HFE T&E and HFE V&V activities, as described in Subsection 18.3.4 and Section 18.4, respectively.

#### **18.3.2.1 Task-Related Input**

A primary input to HSI development is the user task information and control needs established during TA (see Subsection 18.2.4). TA provides the following information that forms the HSI Task Support Inventory:

- Information determining the need to initiate a task
- Control needs to accomplish the task steps
- Information feedback to confirm that task step control actions have been accomplished
- Information for determining that task steps are accomplishing their intended objectives
- Information for determining when tasks may be terminated
- System and component alarms
- Information on task performance requirements for group-use and aggregate HSIs
- Information regarding where manual tasks need to be performed (remote or local)

#### **18.3.2.2 Design Requirements and Guidance Input**

The second main input to HSI design is the full set of HFE design requirements derived from international codes, standards, regulations, and best-practice guidance, that are applicable to the defined user group characteristics and the types of HSIs used throughout the plant. These requirements are managed through the formal requirements management process for the plant design that allows traceability from source to implementation. The HFE Design Requirements Document provides an extract from the requirements management database, providing a single repository of these common requirements (i.e., applicable to all SSC).

For the design of HSIs, the requirements in the HFE Design Requirements Document do not provide the entire basis for developing display interfaces. For example, within “requirement-compliant” screen designs, there are any number of acceptable ways to layout and create screen artefacts, for example, variations in colour, size, font, and placement. Further inputs are required that ensure consistent and intuitive HSI designs across the plant. The requirements for this part of the HSI design process are defined in the project HSI Style Guide.

For digital software-driven HSIs, the style conventions are further developed into an HSI Element Library, which contains HSI display templates and HSI elements (e.g., symbols, numerical displays, graphs) that the display designer uses to assemble the display content. The HSI

Element Library contains both HSI elements for primary interfaces (those that represent direct interface to the system and plant HSI) as well as secondary interfaces (such as navigation, which do not directly relate to system equipment).

The HSI Style Guide is also used to maintain consistency for hardware-based controls and indicators, where suitable components are selected and, along with HSI panel templates, their specifications are included in the HSI Element Library.

#### **18.3.2.3 Human Factors Engineering Concept of Operations**

The HSI concept design scope includes development of the HFE Concept of Operations (COO) which defines the physical and cognitive characteristics of the standardized plant full user population. The HFE COO provides user population anthropometrics for the full range of 5<sup>th</sup> percentile female to 95<sup>th</sup> percentile male users, for the worldwide population specified in ISO 7250:3. Other user population characteristics are provided, including population stereotypes (i.e., expectations of interface functionality of the whole user population based on country or nuclear industry norms).

#### **18.3.2.4 Human Factors Engineering Support**

The final input is provided by the HFE team members on an as-required basis. This input is specific to each design challenge or designer technical query. The integration of the HFE team with the other disciplines provides the mechanism for designers to request HF Engineer/HFE Specialist support for instances where is not clear, for the HSI design aspect they are implementing, how the pre-specified requirements are correctly or effectively applied. Designers also request support when they identify conflicting design criteria that limit or prevent implementing the HFE design requirements as specified. In such cases, they need HF Engineer/HFE Specialist advice on the most suitable alternative design solutions.

#### **18.3.2.5 Results from Testing, Evaluation, Verification and Validation**

Throughout the design development, HFE T&E is performed (Subsection 18.3.4). Later in detailed design, early HFE V&V activities start. The results from these HFE T&E and V&V activities may be the identification of an HFE issue with the design or an HED. Recommended resolutions requiring HSI design improvement form the inputs to further design development.

### **18.3.3 Human-System Interface: Detailed Design and Integration**

In accordance with the HFEPP and the HSI design methodology, HSI designs are created through the interaction and coordination of the HFE team and discipline engineers. Degree and type of interaction is based on the risk-based HFE Application Level as described in Subsection 18.1.4.2.

#### **18.3.3.1 Objectives and Scope**

The objectives of the HSI design process are to:

1. Translate codes and standards, as well as functional and task requirements, into HSI characteristics, displays, software, and hardware that enhance safety and reduce the risk of human error to as low as reasonably achievable through design
2. Support the principal objectives of the HSI design to provide the indications, controls, and status displays necessary for tasks allocated to each user, for all the required plant functions during all plant conditions, and to provide the user with accurate, complete, and timely information regarding the functional status of plant equipment and systems
3. Ensure design trade-offs are resolved during the HSI design activities through the systematic application of HFE principles and criteria, and with HF Engineer/HFE Specialist support

4. Maximize the plant capacity factor in the HSI design by:
  - a. Facilitating planned operations, maintenance, inspection, and testing
  - b. Minimizing the occurrence of any undesired power reduction or plant trip caused by erroneous decision-making and actions
  - c. Permitting plant commissioning to take place effectively and allowing timely modifications and maintenance of the HSIs

The scope of the HSI design process is to specify requirements for HSIs throughout the plant. For hardware-based HSIs, the HFE team supports other discipline designers in designing and selecting HSIs. In the case of software-based HSIs, developing the optimized displays are HFE team responsibility. As with all other HFE Program activities, the scope and methods used for HSI design are graded based on HFE Application Level (Subsection 18.1.4.2). The HFEPP details the level of effort and scope for HSI design per HFE Application Level.

#### **18.3.3.2 Methodology**

The HSI design products are created through the interaction and coordination of the HFE team and discipline engineers. Degree and type of interaction is based on the HFE Application Level.

The HFE team provides design and task support requirements (Subsection 18.3.2).

Depending on the HFE Application Level and the nature of the HSI, the HFE team:

1. Provides design requirement-compliant, application-specific wireframes, and templates for HSI displays, panel layouts, and HSI elements, housed within the HSI Element Library
2. Works with the System Engineer to implement the HFE requirements in the requirements management database (as compiled in the HFE Design Requirements Document) that apply to the discipline and system, allowing them to design or specify and select compliant SSC or HSIs and to develop compliant system and equipment layouts
3. Depending on HFE Application Level, reviews, tests, and verifies all HSI design work to ensure acceptable requirements are implemented and compliance is documented or performs proportionate design work audits using the Design and Task Support Evaluation Checklists to ensure acceptable requirements are implemented and compliance is documented

Depending on the HSI type and the HFE Application Level, the same process is followed but the primary and secondary designers/engineers may vary.

The HFE Program HSI Design technical element includes roll-out of the HFE requirements and support to their implementation for other disciplines. For hardware HSIs, including plant layout and physical environment, direct physical SSC interfaces and HSIs that form part of COTS or bespoke design systems and equipment, the responsible discipline engineer includes the applicable task-based and HFE Design Requirements Document requirements as part of their system and component level design requirements. The applicable HFE requirements are included in "lower" level system and component requirements specifications, including procurement specifications, ensuring consistency of application throughout the plant design. This is managed using the design requirements management tool, the standard content for system design specifications, and the integrated HFE design support activities and issues management process outlined in HFEPP.

The HSI design process for software-based HSI display designs, which are the responsibility of the HFE team, is to create each system User Interface Specification (UIS), which contains a DCT.



NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

The DCT lists the I/O points associated with a HSI element on a display or panel and provides a mapping of instrumented parameters and controlled components to individual HSIs as follows:

1. Assemble inputs and requirements following the UIS input gathering procedure
2. Complete the system UIS and the DCT. The UIS provides detailed renderings of the HSI display and panel layouts, including the components and parameters to be included on the HSI displays and panels. A UIS is created for each system, and a UIS is also created for the plant-level displays as a part of control room system design.
3. Integrate the UIS and DCT for each system

The complete UIS standard deliverable provides the data, templates, and formats necessary for the development of the software and I/O points to drive the user display interfaces. The UIS standard deliverable consists of:

- HSI Task Support Inventory
- HSI Task Support Inventory – Key Parameters
- DCT
- HSI display screenshot

The HSI display screenshot is a record of the actual HSI display, contained in a software file, delivered with the UIS to the I&C team for data connection with the logic modules.

The HSI design process also includes development of an Alarm Management Design Guide during the concept design stage. The guide provides detailed alarm system and alarm presentation guidance, including outlining the principles for alarm identification, prioritization, filtering, and suppression, in line with human cognitive capacity and required response.

For HSI design, applicable requirements are applied to the developing design, and compliance is documented and maintained by the relevant engineering discipline. Where exception to a requirement is needed, HFE provides design support to the other disciplines to develop and document an HFE-approved justification for an HFE design requirement exception.

The HFE design requirements apply equally to the HSIs of COTS equipment and components. Ability of COTS equipment and component HSIs to meet the HFE design requirements is one of the standard selection criteria. However, it is recognized that not all COTS items require the same level of rigour; standard items that do not include HFE as part of their specification require HFE evaluation for non-compliance with the HFE design requirements. When evaluating COTS products that do not comply with HFE design requirements, special considerations are applied by the HFE team to determine and document acceptability of the discrepancy; these include:

1. Trade-off of benefits of using a proven, standard solution compared to the benefits of a custom solution that more closely meets the HFE design requirements
2. Analysis of COTS vendor HFE design basis and documentation in relation to HFE codes, standards, and relevant good practice
3. Evaluation of COTS HSI design applicability to the defined user population, conventions, and stereotypes
4. Degree of design and task support integration and consistency between the COTS product and the rest of the HSIs
5. Identification of usability or human performance concerns with the proposed application of the COTS product

To accommodate these evaluations, where COTS products are considered, this is determined as early as possible in the design lifecycle, following completion of the HFE COO and other HSI design input documents as soon as the relevant System Engineer specifies that such products will be used.

The individual system and integrated plant-level UIs, hardware-based HSI designs and COTS HSIs, and the plant-level HFE requirements (e.g., those related to workspace layout and working environment), are also integrated into relevant control facility HFE design specifications. The design process for the MCR and SCR are described in Subsections 18.3.5 and 18.3.6; the design for other control facilities, including those for responding to accidents and emergencies follows a similar process, with graded level of effort based on complexity of the HSIs. The facilities for supporting emergency and accident response are in early concept design as described in Chapter 7, Section 7.7 and Chapter 19, Section 19.2.

#### **18.3.3.3 Results**

The process and the rationale for the HSI design are documented and managed under GEH Quality Assurance and BWRX-300 specific design plans, as described in Chapter 17. The HSI design process uses the following templates within the TA and HSI Design workbook to create the UIS:

- HSI Task Support Inventory
- HSI Task Support Inventory Key Parameters
- DCT

The specific controls, indications, displays, panels, and HSI elements designed to support the user tasks are documented in the HSI Task Support Inventory table. Data in the table documents that the designer confirmed that the HSI characteristics are appropriate for the specific use application. The table looks at display and panel locations to confirm that information that needs comparison is located on the same display or panel and that information used in related task actions are located on the same display or panel or are available for concurrent display on adjacent video display units or panels.

In addition to the templates, a screenshot of the resulting HSI display is included in the workbook. The HSI design for displays performed by the HFE team also results in the HSI display software file.

To support the standard plant alarm system logic implementation, a detailed alarm presentation specification is produced in line with the Alarm Management Design Guide. The specification includes rationalization and prioritization evaluation results, providing alarm filtering and presentation requirements as input to the I&C team alarm system design activities.

Other outputs include the design documentation (system design specifications, system and component requirements documents, purchase specifications, and drawings and models), as appropriate to the discipline and HSI type being designed. The results of any non-compliance evaluations and design trade-off decisions are recorded in design records and (where appropriate) the HFEITS (Subsection 18.1.5).

#### **18.3.4 Human-System Interface: Tests and Evaluations**

T&E is an integral part of the HFE design process, with the results of evaluation T&E efforts leading to early and effective modification to requirements and design improvements.

#### **18.3.4.1 Objectives and Scope**

The purpose of HFE T&E is to find and address issues early, rather than waiting for HFE V&V activities near the end of the project (Section 18.4). It is the means to test the feasibility of concepts and early prototypes and to facilitate reaching design decisions. Another difference from the V&V is that design and HFE engineers involved during the design stages are not excluded from being test participants.

The scope of the HFE T&E includes:

- Defining the HSI prototypes and simulation testbeds
- Defining the HFE T&E team and participants
- Establishing HFE T&E methods
- Performing HSI selection and prioritization
- Performing HSI evaluation and user-based testing
- Collecting and analyzing data
- Documenting results, and communicating them to the relevant stakeholders

HFE T&E scope ranges in complexity from simple user questionnaire responses and comments to empirical, performance-based techniques to assess how the user responds to the design under increasingly realistic conditions. The level and complexity of HFE T&E is based on design phase, task complexity, integration of the design feature to be assessed, and design and project risk (new HSI, new systems, high HFE risk grading).

#### **18.3.4.2 Methodology**

To maximize the effectiveness of HFE T&E, HSIs are selected based on prioritization criteria. Primarily selection and prioritization of the HSIs are based on the HFE Application Level (Subsection 18.1.4.2). Where HSIs support more than one task, which means they may have more than one associated HFE Application Level, the worst-case (highest risk) level is used.

Beyond this grading, additional HSI are selected for HFE T&E inclusion based on consideration of any HSI design assumptions that require T&E. Assumptions made during the design phase are identified and refined so that they are specific enough for testing. The design assumptions are weighted to determine test priority (similar to the grading of human actions based on risk). The T&E focuses on the assumptions that have the highest impact if incorrect and the shortest time to learning the HSI.

Some examples of candidate HSI design assumptions include the following:

- Colours and status coding (short time to learning; medium impact if false)
- Hardware HSIs basis ergonomic check (short time to learning; high impact if false)
- Safety HSIs (long time to learning; high impact if false)
- HSIs related to the highest risk-level graded tasks (long time to learning high impact if false)
- New system functionality (long time to learning; high impact if false)

The T&E program is comprised of multiple assessment methods, with the most dominant being performance-based testing. Performance-based testing consists of observing users, given a goal to achieve, interacting with a suitable representation of the HSI design. Members of the test team

observe the user's actions without intervening, recording what transpires. Post-test analysis focuses on any difficulties encountered by the user, both qualitatively and quantitatively obtained, depending on test stage and testbed fidelity. The results are used to highlight differences between the design team assumptions in developing the HSI and actual user behaviour when using it, indicating potential human error traps in the design.

During design development, the performance-based testing may be formative in nature. This type of testing allows for quick low-fidelity prototyping and problem resolution. During formative testing the test administrator and user both participate in the test. The administrator may prompt for information on what and why a user is performing actions to understand the thought process as well as to understand better the prototype or conceptual design limitations.

At appropriate points in the detailed design, testing is done with a summative approach. During summative testing the test administrator does not participate to limit test bias. Several users are tested separately to allow assessment of error variances and statistical comparison of test results.

Each performance-based test cycle begins with the development of a test plan that outlines the purpose, equipment needed, design features being tested, test and data collection methods, performance measures and acceptance criteria, as well as any testing material where appropriate. Design features selected for user testing, the test fidelity, user representatives chosen, testbed used, and performance measure(s) and acceptance criteria all depend on the maturity of the design at the stage of the testing.

Considerations for test design include:

1. Availability of plant modeling software and integrated HSI design status
2. Availability and fidelity of a mock-up
3. Availability of control area and equipment 3D modeling
4. Availability of procedures, procedure types, and training material
5. Availability of a sufficiently diverse participant population pool that is representative of the user population to the level required for the testing stage
6. Develop an observation and evaluation plan. The observation plan includes written test plans, scripts for observers and evaluators, standardized training for participants, and the same observers or proctors for all runs of an evaluation (whenever possible).

Once the test plan is complete and the testbed selected or designed, the T&E team develop an observation and evaluation plan. The observation plan includes written test plans, scripts for observers and evaluators, standardized training for participants, and the same observers or proctors for all runs of an evaluation whenever possible.

The test is then conducted in accordance with the plans.

The HFE T&E program uses a variety of methods and tools for analyses, reviews, and evaluations of the HFE T&E performed throughout the design process.

Data collection methods are selected appropriate to the type of test or evaluation being conducted, as detailed in the T&E plan. Techniques appropriate for the evaluation of HSI include:

- Participant questionnaires and interviews
- Direct observation of user behaviours (e.g., task time, task errors, HSI interaction or navigation errors)
- Simulation instructor console data

The following criteria are used to select the data collection methods:

- Safety and risk significance
- Type of design (depending on the type of design, there are some methods that may not apply)
- Type of technology
- Relative time to perform the test or evaluation
- Relative complexity
- Relative cost and value

In addition to performance-based user testing, the HFE T&E team conducts formal trade-off evaluations to determine the relative benefits of potential design alternatives. Trade-off evaluations are conducted by a multi-discipline group of relevant stakeholders – HFE, other discipline engineering experts, HSI designers, and samples of end users. The trade-off evaluation is conducted using a standard trade-off tool, ranking the design alternatives against weighted key HFE criteria. The output of the trade-off tool is used as the basis to make the HSI design alternative trade-off decision. If there are several closely ranking alternatives, further HFE review or analysis is undertaken to determine. HFE issues resulting from this evaluation are recorded and tracked using the HFEITS. Those that are not resolved at the time include the necessary information to address them in future project stages.

#### **18.3.4.3 Results**

The results from any HFE T&E are documented in design records and on associated test forms design to support the T&E process. Any issues and recommendations resulting from the HFE T&E activities related to design improvement are communicated directly to the applicable HSI designer, used as input to further design development (as per Subsection 18.3.2.5). When the T&E for each design stage is complete, a T&E summary report is prepared that summarizes all T&E activities and their results).

#### **18.3.5 Human-System Interface: Design of the Main Control Room**

The current concept of the MCR is described in Chapter 7, Section 7.5. The MCR design features are based upon proven technologies and are demonstrated, through broad scope control room dynamic simulation during HFE T&E and V&V, to satisfy the HSI design goals and design bases. Validation of the implemented MCR design includes evaluation of the design features, the user job roles, staff complement, and procedures, performed as part of the HFE V&V process as defined by the test specification and performance measures specified for each validation activity (Section 18.4).

The HSI design implementation activities include support to the development of dynamic models for evaluating the overall plant response as well as individual control systems, including operator actions. These dynamic models are used to:

- Analyze both steady state and transient behaviours
- Confirm the design of the advanced alarm system concepts
- Confirm the adequacy of control schemes
- Confirm the allocation of control to a system or an operator
- Develop and validate system and plant-level operating procedures

Using part-task simulation, system models are developed, and linked to the HFE-designed the HSI displays. The part-task simulator is used in preliminary plant design and includes design features specific to BWRX-300.

As the design progresses, the part-task simulator proceeds through a series of iterative evaluations resulting in the development of a full-scope simulator. As soon as available, simulators are the preferred testbed for T&E, allowing for progression from static to dynamic testing.

Safety margins used in the DSA account for uncertainty and provide an added margin to ensure that the various limits or criteria important to safety are not challenged. Suitable margin is also added to the HFE analysis of human actions (during TA and human error analysis, per Subsections 18.2.4 and 18.2.6) by ensuring suitable conservatism is included in things like the timeline analysis or the generation of human error probabilities.

Design goals and design bases for the design of HSIs in the MCR and SCR and in other applicable facilities are established in Subsection 18.3.1, based on the HFEPP.

#### **18.3.5.1 Objectives and Scope**

The primary goal of HSI design and HFE input to design of the MCR is to facilitate safe, efficient, and reliable user performance during all phases of normal plant operation, abnormal events, and accident conditions. To achieve this goal, information displays, controls and other interface devices in the control rooms and other plant areas are designed and implemented in a manner consistent with best HFE practices. Further, the following specific design bases are adopted:

1. HSI design promotes efficient and reliable operation through application of automated operation capabilities.
2. HSI design uses only proven technology.
3. Safety-related systems monitoring, and control capability is provided in full compliance with regulations regarding divisional separation, and independence.
4. HSI design is highly reliable and provides functional redundancy such that sufficient displays and controls are available in the MCR, or as a backup, in the SCR and remote locations to conduct a reactor shutdown and to ensure the reactor achieves and maintains safe shutdown conditions, even during postulated accidents.
5. The MCR remains habitable and protected for all events and accidents during which it is required to be used (habitability of the MCR is described in Chapter 6).
6. The principal functions of the SPDS as required by CNSC REGDOC-2.5.2 (Reference 18.3-1) are integrated into the HSI design.
7. Accepted HFE principles and methods are used for integrating HFE into the MCR design, in accordance with international best practice and meeting the requirements of CNSC REGDOC-2.5.1 (Reference 18.3-2).
8. HFE design requirements are based on international standards and applicable CNSC regulatory requirements, as outlined in the HFEPP.
9. The principal functions of the SPDS as required by CNSC REGDOC-2.5.2 (Reference 18.3-1) are integrated into the HSI design.
10. The design basis for accident and emergency control and monitoring facilities meets international standards as well as CNSC REGDOC-2.5.2 (Reference 18.3-1), and CNSC REGDOC-2.10.1 (Reference 18.3-3).

The evaluation of the integrated MCR design provides confirmation that the MCR HSIs and other design features are compliant with the HFE Design Requirements Document and any analysis-based design requirements. Refer to Chapter 7 for description of I&C system content and to Chapter 13 for the Conduct of Operations.

#### **18.3.5.2 Methodology**

The MCR concept outlined in Chapter 7 contains a group of workspaces and individual HSIs, which form the foundation for the detailed HSI design. The development of the MCR workspaces and HSI design features is accomplished through:

- Consideration of existing control room OE
- Review of trends in control room designs and existing control room data presentation methods
- Evaluation of modern HSI technologies, including alarm system design, particularly alarm reduction and presentation methods
- Application of relevant compiled requirements from the HFE Design Requirements Document
- Design or specification and selection of individual HSIs
- Specification of the integrated HFE design requirements for the MCR as a whole
- Testing of a dynamic MCR prototype (full-scope simulator)

Detailed task performance criteria are specified as part of the TA (Subsection 18.2.4) and qualitative human error analysis (Subsection 18.2.6). These criteria are used to govern and direct all plant control room designs. These detailed task performance criteria, along with requirements specified in HFE standards and codes, encompass the set of necessary and sufficient design requirements that maintain the implemented plant control room designs in compliance with accepted HFE principles. This includes ensuring that any HSIs required to provide manual backup control to safety systems are identified and provided in a location and using technology (e.g., hardware-based high-reliability controls and displays) that are available in the postulated task conditions.

The full-scope simulator is evaluated under normal and abnormal reactor operating conditions by participants suitably representative of the defined user population, as described in the HFE T&E process in Subsection 18.3.4. Following the completion of the HFE T&E and V&V on the full-scope simulator, the MCR workspace, and HSI design features are finalized.

#### **18.3.5.3 Results**

The results from design MCR design activities are the same as those for the overarching HSI design process as described in Subsection 18.3.3.3. This includes the outputs from the related HFE T&E and V&V activities, and identification and tracking of design-related HFE issues and HEDs.

#### **18.3.6 Human-System Interface: Design of the Secondary Control Room**

The SCR provides means to safely shut down the plant from outside the MCR in a location that is protected and not impacted by the same scenarios that makes evacuation of the MCR necessary. The SCR provides the HSIs for the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain safe shutdown, if the MCR becomes uninhabitable. The SCR is in early concept at the time of issuing this PSAR. The current concept

is described in Chapter 7, Section 7.6. Habitability of the SCR and protection of the route between the MCR and SCR is described in Chapter 6, Section 6.4.

#### **18.3.6.1 Objectives and Scope**

The SCR provides means to safely shut down the plant from outside the MCR in a location that is protected and not impacted by the same scenarios that makes evacuation of the MCR necessary. The SCR provides the HSIs for the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain safe shutdown, if the MCR becomes uninhabitable. The SCR is in early concept at the time of issuing this PSAR. The current concept is described in Chapter 7, Section 7.6. Habitability of the SCR and protection of the route between the MCR and SCR is described in Chapter 6.

#### **18.3.6.2 Methodology**

The methodology for design of the SCR is the same as that for the MCR (Subsection 18.3.5.2) and more generally for HSI design (Subsection 18.3.3.2). As with all HFE Program activities, a proportionate, graded approach is taken to the design of the SCR. Due to the nature and purpose of the SCR and the plant conditions expected when it needs to be used, the human actions are by default important to safety and if incorrectly performed, lead to significant consequences and as such receive the highest HFE Application Level.

#### **18.3.6.3 Results**

The results are captured in the same means as for the MCR design (Subsection 18.3.5.3).

### **18.3.7 Procedure Development**

Procedure development for the BWRX-300 is performed by the HFE team in accordance with a plan that details the inputs, method, and scope of procedure development activities.

#### **18.3.7.1 Objectives and Scope**

The objective of procedure development is to apply HFE principles and guidance to the development of procedures such that they are technically accurate, comprehensive, explicit, easy to use, and validated. The process for procedure development follows applicable requirements from IAEA-TECDOC-1058, "Good Practices with Respect to the Development and Use of Nuclear Power Plant Procedures" (Reference 18.3-4).

The plant procedures are developed as an integral part of the HSI design development. The procedures are developed either as new or modified from predecessor plants. Existing procedures are modified to reflect the characteristics and functions of the plant task types, modes and conditions, and any applicable OE related to procedure design.

The HFE Program includes activities to verify that all functions and tasks assigned to the plant personnel are included in the procedures. The HFE V&V activities include validation of the procedures using the mock-ups, part-task, and full-scope simulators to confirm their usability and accuracy. Procedure development is iterative and progressive, in line with the developing design and results from progressive HFE analyses.

The scope of procedure development addresses all tasks required to meet functional goals in operations, maintenance, inspection, testing, and accident management of the plant. The scope includes development of:

- Procedures Writer's Guides
- Plant and System Operations Procedures (start-up, normal (at power operations), and shutdown)



- Maintenance, Inspection, Testing and Surveillance Procedures (including Refuelling and Outage Planning Procedures)
- Alarm Response Procedures
- Abnormal Operating Procedures
- Emergency Procedure Guidelines for Emergency Operating Procedures (EOP) development
- (EOPs)
- Severe Accident Management Guidelines
- Emergency Mitigating Equipment Guidelines

The procedure development process does not include specific requirements for the development or procurement and implementation of a Computer-Based Procedures (CBPs) platform or tool. However, the Procedure Writer's Guide specifies basic HFE requirements for usability of the CBP interface (the same as any other user interface) and the development of the content and format for any CBP inputs follows the same requirements for each procedure type given in the above list.

#### **18.3.7.2 Methodology**

The procedures development methodology establishes the process for developing technical procedures that are complete, accurate, consistent, and easy to understand and follow.

For each procedure type, a Procedure Writer's Guide is established. The Procedure Writer's Guide establishes objective criteria so that the procedures developed with it are consistent in organization, style, and content. The Procedure Writer's Guide provides instructions for procedure content and format, writing of steps, and specifying lists of terms used.

Procedures are then written after the associated TAs are finished. TA is an iterative process due to the amount of information that is created at any point in the design process. TA is conducted in a prioritized manner and done on a per task, per system basis. System Design Descriptions are used along with the TA output to give a complete understanding of the system and its operation. These inputs provide understanding of each system, its associated tasks, and its interrelationship to other systems, which allows the development of system level and then integrated operations procedures.

As the TA progresses beyond operations, when the System Engineers have defined test, inspection, and maintenance requirements and the iterative safety analyses have sufficiently matured, procedures are developed for other than normal operations (e.g., test, maintenance, surveillance, alarm response, outages and any other conditions that are not included in the scope of normal plant operations). Procedures that support important human actions for the higher HFE Application Levels are developed by the HFE team based on the detailed TA and qualitative human error analysis. Procedures to support human actions for the lowest HFE Application Level are developed by the responsible System Engineer or the vendor, based on the Basic TA (Subsection 18.2.4).

Initial procedures are tested and evaluated for their usability and efficacy using physical mock-ups, simulation, and plant 3D models early in the design, through the HFE T&E activities. The HSI design and procedures are evaluated together to make a more cohesive model of the future operational plant. This provides contextual feedback for both the HSI design and the procedure, allowing optimization of the design of both simultaneously.

The developed procedures are verified and validated as part of the HFE V&V program (Section 18.4), culminating in ISV. Procedures that support important human actions for the higher HFE Application Levels are validated by individuals that are independent of the design. Individuals that participate in validation of these levels of procedures include representatives of the end users. Final procedure validation is done on the installed physical plant hardware as part of the HFE design implementation once the plant has been built. Procedures to support human actions for the lowest HFE Application Level are validated by the designer.

The final procedure validations are done as specified by the plant pre-operational testing and start-up testing programs. Following ISV, the procedures are used as the basis for pre-operational testing, start-up testing, and operation of the plant. Once a procedure is validated and declared complete, procedure maintenance and control of updates is governed by the engineering change management process ensuring that changes to individual procedures are reflected throughout the full suite of related procedures, and that changes are confirmed as accurate and supported by the design and TA. Validated procedures are provided to operations for training and use.

#### **18.3.7.3 Results**

The results of the procedure development process are the final set of procedures and any procedure support documentation developed using the procedure development methodology.

The output documents include the following:

- Procedure Writer's Guides
- Plant and system operations procedures
- Maintenance, Inspection, Testing and Surveillance Procedures (including Refuelling and Outage Planning Procedures)
- Alarm Response Procedures
- Abnormal Operating Procedures
- EPGs
- EOPs
- Severe Accident Management Guidelines
- Emergency Mitigating Equipment Guidelines

#### **18.3.8 Training and Qualification Program Development**

Training and qualification program development is coordinated with the other elements of the HFE Program, for example by using HFE TAs to conduct a systematic analysis of job and task requirements. The program of analysis and training material development is conducted by the Training team, in accordance with a plan that provides the methods and framework for ensuring the program meets its requirements and technical basis. The HFE team provide inputs to the training analysis activities and provide support to the Training team in conducting the training program development activities.

##### **18.3.8.1 Objectives and Scope**

The aim of the BWRX-300 training and qualification program development is to systematically incorporate information from the other HFE design tasks to support development of accurate and applicable training content and implementation of effective personnel training. The training program development process is intended to produce a program that:

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- Identifies all performance requirements of a job or duty area relating to licenced activities
- Defines and documents the training based on a TA that provides the information to establish the knowledge, skills, and abilities to perform each task including identifying the safety-related attributes if any
- Ensures that the training is designed, developed, and implemented to meet the qualification requirements
- Ensures instructors meet and maintain documented qualification requirements, particularly in areas of subject matter expertise and instructional skills
- Ensures that formal evaluation methods are used to confirm and document workers qualifications
- Implements a change management control system that systematically identifies changes to tasks and task lists for revisions of training
- Ensures continuing training is provided as deemed necessary through training needs analysis
- Evaluates training regularly and incorporate the results of the evaluation into a training improvement process
- Ensures that workers training and qualifications records are established and maintained
- Ensures that workers have a level of training related to nuclear safety corresponding to their position including but not limited to radiation safety, fire safety, onsite emergency training, and conventional health and safety

The training and qualification program development includes the following stages:

- Analysis
- Design
- Development
- Implementation
- Evaluation

The overall scope of the resulting training and qualification program includes the following:

1. All categories of personnel conducting tasks within the plant, including the full range of job roles whose actions may affect plant safety
2. The full range of plant conditions (normal operational, outage, abnormal, accident, and emergency)
3. All activities conducted throughout the plant (e.g., operations, radwaste processing, outage refuelling, online and offline maintenance, testing, and inspection)
4. The full range of plant functions and systems
5. The full range of relevant HSIs

The scope of the training and qualification program development plan does not include the specific requirements for certification of plant personnel specified in CNSC REGDOC-2.2.3, "Personnel Certification, Volume III: Certification of Reactor Facility Workers" (Reference 18.3-5). The requirements for certification are incorporated as part of the overall program; however, they

do not need to be derived through the defined development process, since they are already specifically defined. The final training and qualification program specific to the operational goals of the plant and developed from the HFE and design inputs, as described in this section, are augmented by the general and specific training and certification requirements specified in CNSC REGDOC-2.2.3, Volume III (Reference 18.3-5).

#### **18.3.8.2 Methodology**

The training and qualification program development follows the fundamentals of the systematic approach to training method. The development process complies with the requirements of CNSC REGDOC-2.2.2, "Personnel Training" (Reference 18.3-6).

The Analysis Stage provides the identification of training needs, tasks, or competencies required for training and the knowledge, skills, and abilities required to perform a specified job position based on assigned tasks. Tasks that support plant functions are identified as part of Detailed and Basic TA, described in Subsection 18.2.4. Tasks are selected for training based on difficulty, importance, and frequency analysis. Depending on the difficulty, importance, and frequency ranking, a decision is made to determine if initial training and periodic retraining is needed. This evaluation of training tasks is the training equivalent to grading human actions (Subsection 18.1.4.2). The results of the TA, including identification of critical steps, inform the difficulty, importance, and frequency analysis and resulting rankings.

Depending on the difficulty, importance, and frequency ranking, a determination is made if initial and periodic retraining is required. Tasks that are selected for training are then analyzed to determine the required knowledge, skills, and attributes. The knowledge, skills, and abilities necessary for each job position, including entry-level education, training, and experience, is established to support training design. Any changes to the iterative HFE or system design inputs to the analysis phase are required to be assessed for impact on the training analysis.

During the Design Stage, learning objectives are developed and a description of the plan for training, including purposed methods and settings, is established. Specifically, the Design Stage includes the following activities:

- Determine the scope, purpose, and timeframe of the training
- Determine the ideal training environment
- Select training methods and instructional strategies in accordance with the environment
- Determine and group the job role knowledge, skill, and attributes addressed by each training module
- Determine the final and partial learning objectives for each training module, including defining performance statements, conditions statements and performance standards
- Prepare the table of contents and scope for each training module; scope includes number and type of documents developed in the next phase
- Prepare master training procedures and formats to ensure consistency across the course materials
- Prepare the training plans for each job position; plans comprise learning objectives, contents, learning activities, training equipment, and a list of materials needed for training, including guidance for their use

The completion of the Design Stage establishes the input that is needed for the Development Stage.

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

In the Development Stage, detailed lesson plans and instructional materials are created, including any on-the-job training documents, and knowledge and performance assessment tests are established. The materials developed must incorporate the required features specified in the Design Stage. The materials are developed such that they have the following attributes:

- Course material content supports mastery of the subject learning objectives.
- Course materials are structured to provide consistent presentation.
- Course material presentation sequence supports effective learning.
- Course materials support successful presentation in the specified venue(s) the course is to be provided.
- Instructor certifications and training required to present training is specified for each course and supports successful presentation in the required venue. Instructors are trained during this phase.
- Exam question banks and examination structure and content are developed to adequately evaluate, and document trainee mastery of the course and job performance objectives associated with the training.

At the end of the Development Stage, the training package is reviewed, piloted on trainees, and revised if necessary.

In the Implementation Stage, instructors prepare for and deliver the training. Trainees are tested to determine if they have mastered the objectives. The results of trainee tests are examined during the Evaluation Stage. The Evaluation Stage examines the effectiveness of the training as delivered. This appraisal is done through the review of training results, training feedback, and continual monitoring of work performance (Section 18.6).

The training and qualification program, developed as discussed above, provides assurance that plant personnel have the capability and competence needed to perform their roles and responsibilities. Participants used for ISV, as described in Section 18.5, are trained using this program and provide validation of the integrated design.

#### **18.3.8.3 Results**

The specific program outputs include documentation defining the overall program goals and course structure, as well as the specific job role qualification and training requirements and developed course materials. In addition to the training program content itself, the results of the training and qualification program development are summarized in a report which documents the process and activities used in development, including any inputs used, issues identified, and recommendations made.

#### **18.3.9 References**

- 18.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.3-2 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.3-3 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 18.3-4 IAEA-TECDOC-1058, "Good Practices with Respect to the Development and Use of Nuclear Power Plant Procedures," International Atomic Energy Association.

NEDO-33968 REVISION 0  
NON-PROPRIETARY INFORMATION

- 18.3-5 CNSC Regulatory Document REGDOC-2.2.3 "Personnel Certification, Volume III: Certification of Reactor Facility Workers."
- 18.3-6 CNSC Regulatory Document REGDOC-2.2.2, "Personnel Training."

## **18.4 Human Factors Verification and Validation**

HFE V&V is a critical HFE design assurance activity applied to the realized design of the plant HSIs and the working environment where those HSIs are used. The HFE V&V program evaluates the plant design (in parts and as an integrated whole) against HFE design principles and requirements, user task requirements, job design and staff complement, procedural accuracy and usability, and effectiveness of training.

### **18.4.1 Objectives and Scope**

The HFE Verification is conducted through two activities with the following objectives:

1. Task Support Verification (TSV) verifies that the HSIs, as defined and baselined in the HSI inventory and characterization, include the necessary features (e.g., controls, information displays, and alarms) required to support tasks and that there are no unnecessary features.
2. HFE Design Verification verifies that the HSIs and plant SSC, are compliant with the applicable HFE design requirements contained in the HFE Design Requirements Document and design-to-analysis requirements input as a result of HFE analysis activities. Verification activities include identifying changes to the design that impact HSIs and other features due to competing design constraints, and checking for due consideration of OE items, user stakeholder input and HFE T&E results.

HFE Validation is conducted through staged activities, as follows:

1. Early and Partial System Validation activities are performed in advance of the full-scope simulator and fully constructed plant and are generally performed only on partial systems. Although they require a sufficient maturity of the design, HFE participants from the V&V team, and end users with enough level of familiarity with the system, they do not require the full integrated system. The purpose of these validation activities is to identify and solve HFE issues in advance of a fixed design.
2. ISV is the performance-based evaluation of the fully integrated system design. Simulations and virtual reality models are used to validate the ability of personnel, trained using the training and qualification program material, to use the integrated HSIs and finalized procedures in accordance with the task and scenario performance requirements. ISV is intended to evaluate those integrated aspects that were verified and validated singly through earlier, partial means.

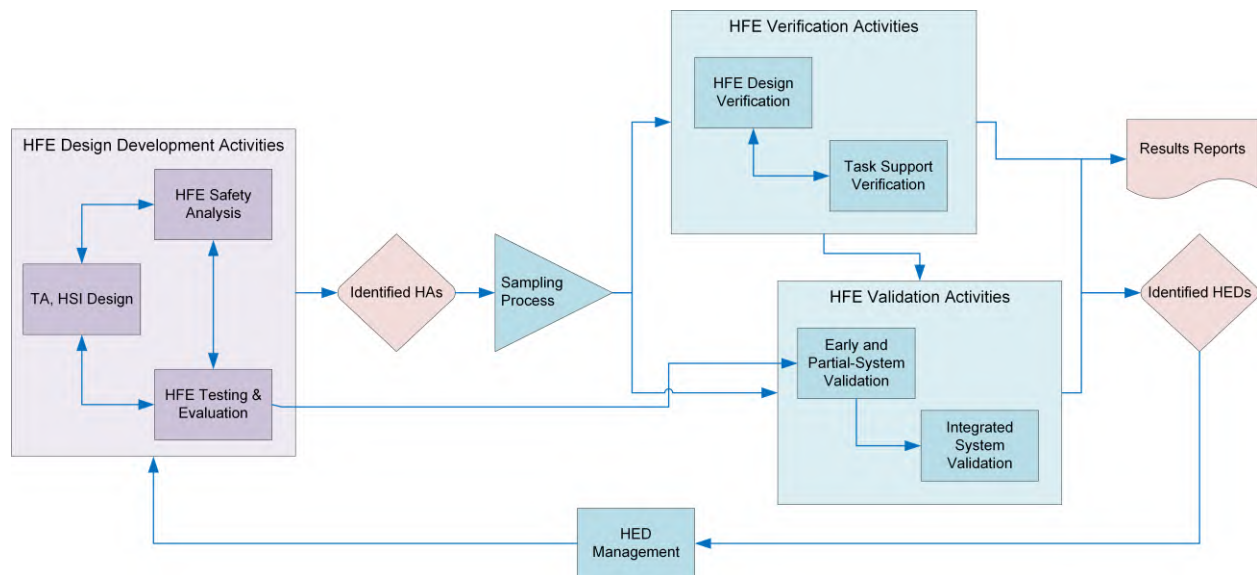
HFE Validation ensures that the design, particularly the HFE-specified aspects, accomplishes its intended goals for usability and reducing the risk of human error to as low as reasonably achievable. Validation is an integrated, dynamic, performance-based test activity in which participants are subjected to a set of simulated scenarios that represent a realistic, challenging, and generalizable set of conditions to ensure that the integrated HSI supports safe operation of the plant.

The scope of the HFE V&V activities applies to user interactions with the plant when performing operations, maintenance, testing, and inspection activities. The HFE V&V activities are applied to HSIs within scope of the HFE Program. As with the other HFE Program activities, the application of HFE V&V is graded to focus on the HSIs, tasks, and plant conditions that involve important human actions, are complex or novel, or are inherently hazardous. The same risk-based approach described in Subsection 18.1.4.2 is applied to the HFE V&V activities to determine the appropriate scope, rigour, and level of detail for each activity.

## 18.4.2 Methodology

The HFE V&V program is conducted in accordance with a structured, systematic plan. The program was developed to meet the requirements and best-practice guidance specified in CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.4-1), CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.4-2), CSA N290.12-14, "Human factors in design for nuclear power plants" (Reference 18.4-3), IEC 61771, "Nuclear power plants – Main Control Room – Verification and validation of design" (Reference 18.4-4), and IAEA SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants" (Reference 18.4-5). The program adopts a risk-based graded and multi-staged approach to V&V.

The V&V program plan specifies overall process used for HFE V&V, and the scope, inputs, methods, and outputs to be used for each V&V activity. The overall process is shown in Figure 18.4-1.



**Figure 18.4-1: BWRX-300 HFE V&V Process Overview**

The Sampling Process is a support activity that establishes the scope of the HFE V&V activities. In a new plant design, the number of scenarios and HSIs is too large to effectively perform HFE V&V to the same degree on all of them. The purpose of the Sampling Process is to focus on the significant, novel, and complex HSIs and tasks, ensuring a full breadth of HFE V&V scope but removing any duplication, thus improving the efficacy of the HFE V&V activities.

The Sampling Process selects the inputs that bound the scope of the HFE V&V activities. The verification activities target a selection of HSIs (e.g., displays, panel layouts, equipment-mounted controls, and indications) and the validation activities target a selection of scenarios. The goal of sampling is to maximize sample relevance and significance while ensuring that the sample is sufficiently broad and diverse, so that the HFE V&V results are generalizable to the overall population of HSIs and scenarios.

TSV compares the HSI elements (alarm, control, information and equivalent) identified during the detailed analysis of a task to the designed HSIs to ensure that all components needed to safely and efficiently complete the tasks present in the final design. The task support inventory and verification criteria are identified during TA (Subsection 18.2.4).



In HFE Design Verification, various aspects of HSI and plant SSC design are compared to the relevant design requirements specified during the design development (Subsections 18.3.1 through 18.3.6). The aspects verified include:

- Static and dynamic HSI features, including HSI-specific and standardized features
- Interface management features such as navigation and data retrieval
- Workstations and workspace anthropometrics
- Global workspace features (i.e., layout, workplace environment, lighting, noise)
- Effects of degraded HSI and plant workplace conditions

During HFE Design Verification, the HFE verifier documents each HSI, or plant SSC element being evaluated (including document and page numbers, screenshots, or photographs as applicable), which subset of HFE Design Requirements Document requirements were applied, and whether the HSI or SSC element passed or failed each requirement.

Early Validations form an essential part of the HFE Validation activities; they are performed to identify and solve HFE issues in advance of a fixed design. They require a sufficient maturity of the design, HFE participants from the V&V team, and end users with enough level of familiarity with the system. However, they do not require the full integrated system, including trained users and final procedures, that the ISV requires. Early Validations are expected to progress HFE T&E activities and results (Subsection 18.3.4), using higher fidelity testbeds and more cohesive scenario-based sets of tasks. The general method for conducting the Early Validations is the same as that for ISV, without the requirements for a complete integrated system and complex high-fidelity testing environments.

ISV is the performance-based evaluation of the fully integrated system design. Simulations and virtual reality models are used to validate the ability of personnel, trained using the training and qualification program material (Subsection 18.3.8), to use the integrated HSIs and finalized procedures such that they support safe plant functionality. ISV is intended to evaluate those integrated aspects that were verified separately through earlier, partial means (i.e., through HFE T&E, TSV, Early Validation). ISV is performed using high-fidelity simulators, task trainers or virtual reality labs (i.e., for scenarios outside of control rooms and control stations). The ISV is the final activity that ensures the integrated design is fulfilling its intended function and demonstrates that claims made in the safety analyses are achievable to the performance requirements specified.

The general method for conducting either early validations or ISV is:

1. Perform preliminary activities
  - a. Scenario identification and development
  - b. Testbed verification
2. Perform testing
  - a. Participant selection
  - b. Scenario definition and documentation
  - c. Performance measures
  - d. Test design
  - e. Pilot testing
3. Perform data analysis and document results

Any issues or non-compliances identified during the HFE V&V activities are identified as a HED. HEDs are processed using the HFEITS process as described in Subsection 18.1.5.

#### **18.4.3 Results**

The results of all the HFE V&V activities are captured in the following indicative documents:

1. HFE Verification Results Report
2. HFE Early and Partial Validation Result Report(s)
3. HFE ISV Test Specification
4. HFE ISV Summary Report

Another output of the HFE V&V activities is identified HEDs, which are captured in the HFEITS and managed through the process described in Subsection 18.1.5.

#### **18.4.4 References**

- 18.4-1 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.4-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.4-3 CSA N290.12, "Human factors in design for nuclear power plants," CSA Group.
- 18.4-4 IEC 61771, "Nuclear power plants – Main control room – Verification and Validation of Design," International Electrotechnical Commission.
- 18.4-5 IAEA Safety Standards Series No. SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants," International Atomic Energy Association.

## **18.5 Design Implementation**

Design Implementation addresses the implementation of the HFE design requirements in the final realized design.

### **18.5.1 Objectives and Scope**

The HFE Design Implementation activities have the following objectives:

1. Confirm that the final realized HSIs, plant SSC, procedures, and training conform to the design requirements and design documents resulting from the HFE Program activities
2. Identify any deviations from the design during implementation and assess their impact on the HFE aspects of the design
3. Perform final procedure validation on the physical plant hardware
4. Verify aspects of the design that may not have been evaluated previously in the V&V process, i.e., any HSIs that were absent or modified from the simulator-based ISV, or any plant physical or work environment (e.g., noise, lighting, thermal) characteristics; verification of items not previously identified as needing evaluation uses the same grading process as the original verification during the design stage
5. Verify and document the resolution of all remaining HFE issues and HEDs (Subsection 18.1.5)
6. Verify HFE Application Levels (Subsection 18.1.4.2) are correct based on the final version of documents and data used as input to determine the levels

The scope of the design implementation is the full set of HFE aspects of the plant including design of the HSIs and plant SSC, plant procedures, and finalized training documentation.

### **18.5.2 Methodology**

Unlike the other HFE technical elements, Design Implementation is performed after the design is complete, immediately prior to commencement of commercial operations. Despite the plant being built and undergoing start-up testing and commissioning, the HFE team still performs the Design Implementation activities. The process follows the HFE V&V process. In this case, the Sampling Process includes identifying any changes to the standardized plant design (documented through engineering configuration control process) and assessing impact on HFE aspects of the design. The Sampling Process also identifies aspects of the design that were not able to be previously verified and validated. Additional items not previously identified for V&V but determined to require it are also added to the scope for Design Implementation HFE V&V.

The list of methods used are similar to those described for the HFE Program activities, particularly HFE V&V, Staffing Analysis, Procedure Development, Training and Qualification Program Development, and HFE Issues Tracking.

At this stage, the plant design is in formal engineering configuration control, as described in Chapter 17.

### **18.5.3 Results**

The results of the Design Implementation activities match the outputs for the same activities conducted during the design phase. They are recorded in a summary report. Any remaining HFE issues or HEDs are recorded in the HFEITS for turnover to the plant operating organization, and further mitigation through operating arrangements as required.

## **18.6 Human Performance Monitoring**

The HPM strategy links HFE methods used during the design with methods for monitoring user task performance during operation. The HPM program is fully developed by the licence applicant as part of the future licencing stage.

### **18.6.1 Objectives and Scope**

The purposes of HPM are:

1. To ensure that the high safety standards established by the HFE Program during the design of the plant are maintained even when changes are made to the plant
2. To detect any deterioration of task performance that may be attributable to latent or slow-developing HFE design issues
3. To provide adequate assurance that the safety bases remain valid during the operational phase of the plant

There is no intent for the HSI designer or the applicant to periodically repeat a full set of ISV activities. The strategy is to provide a monitoring plan, building upon the HFE activities during the design that can be carried forward into the operational phase, using industry accepted methods. HPM incorporates the monitoring strategy into the problem identification and corrective action program, which identifies and classifies human errors, provide for evaluation of the root cause, and supports effectiveness verification and documentation of the corrective action.

The scope of the performance monitoring strategy provides reasonable assurance that:

1. The HSI design is effective during:
  - a. Normal operations
  - b. Maintenance, Inspection, Testing, and Surveillance
  - c. Anticipated Operational Occurrences
  - d. Design Basis Accidents
  - e. Design Extension Conditions
  - f. Severe Accidents
2. Human actions, using HSI information, cues and controls can accomplish tasks while maintaining margin for time and performance criteria.
3. Acceptable performance levels established during the HFE ISV are maintained.
4. Changes made to the initial HSIs, user group definition, job design, procedures, and training do not have adverse effects on personnel task performance (e.g., a change interferes with trained skills, or a fatigue management policy is not implemented, contrary to what was assumed in the HFE COO).

### **18.6.2 Methodology**

The HPM program aligns with the overall quality program and condition reporting methods. The program includes:

- Data collection
- Importance screening
- Event analysis to determine causes

- Trend analysis
- Corrective action development

The HPM strategy collects data to trend task performance, particularly seeking issues with design root causes due to non-compliance with or inappropriate application of HFE principles. The HPM program uses existing utility or industry programs (e.g., corrective action, programs, or operator training) for data collection where appropriate. The HPM program is designed to ensure that:

1. Human actions are monitored commensurate with their safety importance.
2. Feedback of information and corrective actions are accomplished in a timely manner.
3. Degradation in performance is detected and corrected before plant safety is compromised.

*NOTE: The HFE-based HPM does not seek personnel behaviour-based corrective actions. It is focused solely on issues related to the design of HSIs and organizational arrangements that lead to human error.*

The HPM program maintains a database of event causes and corrective actions taken. Such data supports trending of performance anomalies.

The HPM identifies and establishes corrective actions that reduce the potential for incident recurrence. The program systematically identifies the cause of the failure or degraded performance. The corrective actions are derived by:

1. Addressing the significance of the failure through application of PRA/HRA importance measures
2. Classifying the causes and circumstances surrounding the failure or degraded human performance
3. Illuminating the characteristics of the failure (e.g., being task specific or due to design issues)
4. Determining whether the failure is isolated or has generic or common cause implications

### **18.6.3 Results**

The HPM program activities and outputs align with the overall condition reporting requirements. They are expected to include specific incident or trend analysis reports, a recommendations and action tracking database, and periodic summary reports.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33969

Revision 0

September 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 19  
Emergency Preparedness and Response**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release



### ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
BDBA	Beyond Design Basis Accident
CB	Control Building
CNEP	Consolidated Nuclear Emergency Plan
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
DBA	Design Basis Accident
DEC	Design Extension Condition
DL	Defense Line
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EME	Emergency Mitigating Equipment
EMEG	Emergency Mitigating Equipment Guideline
EOC	Emergency Operations Centre
EOP	Emergency Operating Procedure
ERF	Emergency Response Facility
ERO	Emergency Response Organization
IAEA	International Atomic Energy Agency
LTC	Licence to Construct
MCR	Main Control Room
OPG	Ontario Power Generation
PNERP	Provincial Nuclear Emergency Response Plan
PPE	Personal Protective Equipment
REGDOC	Regulatory Document (Canadian Nuclear Safety Commission)
SAMG	Severe Accident Management Guideline
SCR	Secondary Control Room
SPDS	Safety Parameter Display System
USNRC	U.S. Nuclear Regulatory Commission

## TABLE OF CONTENTS

<b>19.0 EMERGENCY PREPAREDNESS AND RESPONSE .....</b>	<b>19-1</b>
19.1 Arrangements for Performing Functions Essential for the Emergency Response	19-1
19.2 Emergency Response Facilities.....	19-7
19.3 Capability of the Operating Organization for the Assessment of Potential Radioactive Releases in Accident Conditions .....	19-8
19.4 Emergency Preparedness for Multiple Unit Sites .....	19-9
19.5 Emergency Preparedness for Internal Fire Hazards .....	19-10
19.6 References .....	19-11

**LIST OF TABLES**

None.

## **LIST OF FIGURES**

None.

## **19.0 EMERGENCY PREPAREDNESS AND RESPONSE**

The purpose of Chapter 19, Emergency Preparedness and Response, is to provide sufficient information on emergency arrangements, demonstrating in a reasonable manner during a nuclear or radiological emergency, all actions necessary for the protection of workers, the public, and the environment could be taken; and the decision-making process for the implementation of these actions would be timely, disciplined, coordinated, and effective.

To maintain consistency throughout the OPG Nuclear fleet, including the Darlington New Nuclear Project (DNNP), the content of Chapter 19 is based on the current OPG Nuclear emergency management program documented in N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan (CNEP)" (Reference 19.6-1), which is compliant with Canadian Nuclear Safety Commission (CNSC) REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response" (Reference 19.6-2); CSA Group (CSA) N1600, "General requirements for nuclear emergency management programs" (Reference 19.6-3); and "Provincial Nuclear Emergency Response Plan (PNERP): Master Plan" (Reference 19.6-4).

During construction and fuel-out commissioning phases, as defined in CNSC REGDOC-2.3.1, "Conduct of Licensed Activities: Construction and Commissioning Programs" (Reference 19.6-6) of the project, the potential for nuclear and radiological emergencies will be the result of incidents from Darlington Nuclear Generating Station (DNGS) only. Since an event at DNGS, which includes site facilities listed in Chapter 2, Subsection 2.2.5, may affect personnel at DNNP, emergency response protocols are considered for both the DNNP and DNGS sites. Therefore, all personnel at DNNP will appropriately respond to emergency tones and announcements emerging from DNGS. This follows the requirements of NK054-PLAN-01210-00002, "DNNP Nuclear Emergency Preparedness Plan" (Reference 19.6-5).

In support of the Licence to Construct (LTC) application, Chapter 19 complies with CNSC REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility" (Reference 19.6-15), specifically Section 4.10, Emergency Management and Fire Protection and Subsection 4.3.5, Accident and Severe Accident Management and Recovery, CNSC REGDOC-2.3.1 (Reference 19.6-6), specifically Subsection 3.3.5, Emergency Management and Fire Protection, CNSC REGDOC-2.3.2, "Operating Performance – Accident Management" (Reference 19.6-13), and applicable Sections of CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 19.6-8). In addition, other applicable regulatory documents, codes, and standards related to emergency preparedness and response are referred to in the specific discussion topics throughout this chapter.

### **19.1 Arrangements for Performing Functions Essential for the Emergency Response**

The CNEP (Reference 19.6-1) describes concepts, structures, roles, and processes to implement and maintain an effective OPG response in the unlikely event of a nuclear or radiological emergency that could endanger on-site staff, the public, or the environment. The CNEP provides a framework for interaction with external authorities and defines OPG commitments under the PNERP Master Plan (Reference 19.6-4).

The PNERP provides the off-site planning basis for nuclear emergencies with the goal of ensuring public safety in the event of a nuclear emergency. The PNERP Master Plan (Reference 19.6-4) establishes the principles, concepts, organization, responsibilities, policy, functions, and inter-relationships, which govern all off-site nuclear emergency planning, preparation, and response in Ontario.

As part of DNNP Site Preparation Commitment D-P-5.1, OPG developed the DNNP Nuclear Emergency Preparedness Plan (Reference 19.6-5), which takes its program authority from the

NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

CNEP (Reference 19.6-1). The plan provides a written basis to document the concepts, roles, and resources required by OPG to implement and maintain its emergency response on the DNNP site to protect employees, visitors, and contractors in the event of a nuclear emergency originating from DNGS operations. The document describes the different phases of the DNNP Site Preparation, Construction, Commissioning and Operations, and how the nuclear emergency preparedness program will transition from construction to fuel-in commissioning when the site will become a Class I nuclear facility.

Under Commitment D-O-2 (Reference 19.6-6), submitted with the Licence to Operate application, OPG will develop or revise the CNEP (Reference 19.6-1) to address site specific emergency planning for the DNNP including the following:

- The basis for emergency planning
- Selection and qualification of workers
- Emergency preparedness and response organization
- Emergency training
- Drills and exercises
- Emergency procedures
- Assessment of emergency response capability
- Assessment of accidents
- Activation and termination of emergency responses
- Protection of facility workers and equipment
- Interface with off-site organizations
- Recovery program
- Public education program

The nuclear emergency plan for BWRX-300 will comply with CNSC REGDOC-2.10.1 (Reference 19.6-2) and the PNERP Master Plan (Reference 19.6-4).

OPG has a Memorandum of Understanding in place with the Province of Ontario to revise the PNERP prior to fuel-in commissioning, as defined in CNSC REGDOC-2.3.1 (Reference 19.6-6) to include DNNP and issue a revised Darlington Implementing Plan or a separate Implementing Plan for DNNP.

The basis for emergency response is to implement plans and procedures to protect site personnel, the public, the environment, and to minimize adverse effects to structures, systems, and components within the site perimeter and beyond.

The OPG Emergency Response Organization (ERO) is responsible for the following:

- Initial notifications
- Performing site surveys
- Bringing the incident unit under control
- Any immediate personnel protective actions for the site
- Providing data for off-site agencies to make public protection decisions

NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

Some or all of these actions will apply to the BWRX-300 and will be determined as information becomes finalized.

The integration of resources and support from both internal and external parties are essential to ensure on and off-site measures are administered per the CNEP (Reference 19.6-1), PNERP Master Plan (Reference 19.6-4) and CNSC REGDOC-2.10.1 (Reference 19.6-2).

The purpose of the OPG Nuclear emergency management program (Reference 19.6-1) is to implement plans and provide trained personnel to address nuclear events by controlling releases and minimizing effects of radioactive releases. OPG also works with government agencies, regional partners, and other stakeholders to protect the public and plan for recovery following a nuclear emergency.

To remain consistent with OPG Nuclear emergency response practices, similar response processes will be employed at DNNP. The DNNP senior authorized person on shift during a declared emergency performs the following:

1. Assumes command and control of emergency response operations from the outset of the emergency
2. Be located in either the Main Control Room (MCR), Secondary Control Room (SCR) or on-site Emergency Operations Centre (EOC)
3. Fulfills the responsibilities of the emergency response Commander until relieved by the call-in emergency response Commander supported by the augmented ERO
4. Until relieved, executes both the senior shift licence duties and specific responsibilities of the emergency response Commander (e.g., off-site interface role, overall site commander and chief of response)

The DNNP senior authorized person on shift performs an emergency assessment, including classification and categorization, in order to determine the following:

1. Extent of the on-site response and staff mobilization required for the protection of on-site personnel and equipment
2. Notification category necessary for the Province of Ontario to determine the off-site response required for the protection of the public and the environment

To generate the appropriate level of response to an incident, the DNNP senior authorized person on shift must first identify the significance of the incident (D-INS-03490-10009, "Shift Manager" (Reference 19.6-11)). The processes used are termed "Classification" and "Categorization". The actual or potential threat to employee and public safety will determine the event classification and categorization.

Classification determines the level of on-site response, whereas categorization determines the level of off-site response.

The classification depends on in-plant circumstances. Should the event be classified as a Station Emergency, the Station Emergency Tone (D-INS-03490-10007, "Response to Warning Signal" (Reference 19.6-14)) will be sounded (or other established on-site notification protocols).

NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

A Station Emergency is a radiological or non-radiological incident (e.g., fire, explosion, flood, tornado, seismic event, hazardous material, major system/equipment failure, security breach, etc.) where one or more of the following emergency response activities are required, more quickly than could be activated via normal supervisory contact:

- Assembly of personnel to designated safe areas
- Accounting of personnel
- Mobilization of ERO resources to manage the situation

First, the process of OPG emergency classification is completed. Then, a second process is undertaken to determine the Provincial notification category and associated default off-site protective actions. The assessment process for off-site categorization is governed by the criteria set by the PNERP (Reference 19.6-4). OPG has a Memorandum of Understanding with the Province of Ontario to review the PNERP categorizes in order to develop site specific instructions applicable to DNNP.

The Shift ERO will initially assume command and control of the site emergency. Augmented ERO personnel will be called in to form the Site Management Centre staff to support the shift organization. The Shift ERO will transfer coordinated site response to the Site Management Centre when it becomes operational. See Section 19.2 for Emergency Response Facilities description.

The Site Management Centre staff ensures coordinated site response occurs and site commitments under the emergency plan are met. The Site Management Centre staff will assume responsibility for on-site personnel protective actions, coordination of site evacuation, direction of operational recovery, responsibility for communications and interface between the station and the off-site Emergency Response Facility (ERF), as well as with external organizations.

Staff in the ERO, as indicated above, are relieved by replacement staff depending on the duration of the response.

OPG has established plans and procedures to coordinate with the appropriate off-site organizations. These plans and procedures will be used as a framework for development of BWRX-300 site specific procedures per this chapter and Chapter 13, Subsection 13.4.3, Procedures and Guidelines for Operating the Plant During Accidents.

Categorization determines the level of off-site response as defined in the PNERP (Reference 19.6-4) and based on the order of significance as follows:

- Reportable event
- Abnormal incident
- On-site emergency
- General emergency

Details of the notification requirements for the four defined categories are as defined in the PNERP (Reference 19.6-4). See subsequent discussion on notification target times.



NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

As previously stated, on-site emergency arrangements, including programs on training and exercises, will be submitted with the Licence to Operate application under Commitment D-O-2 (Reference 19.6-5). This will ensure emergency preparedness and response is in place prior to fuel-in commissioning, which is within the scope of the Licence to Operate. Additionally, OPG will meet the requirements of CNSC REGDOC-2.3.1 (Reference 19.6-6), which includes testing emergency management provisions before the commencement of fuel loading.

CNSC REGDOC-2.10.1 (Reference 19.6-2) and PNERP (Reference 19.6-4) encompasses several national and international standards and guidelines, including International Atomic Energy Agency (IAEA) Safety Standards Series No. GS-G-2.1, "Arrangements for Preparedness for a Nuclear or Radiological Emergency" (Reference 19.6-7). Additional standards are referenced in other nuclear programs, including those for radiation protection.

Analysis of Design Basis Accidents (DBAs), Design Extension Conditions (DECs) and Beyond Design Basis Accidents (BDBAs), including severe accidents for the BWRX-300, is discussed in Chapter 15, Safety Analysis. The systematic approach to develop the Emergency Operating Procedures (EOPs), Emergency Mitigating Equipment Guidelines (EMEGs) and Severe Accident Management Guidelines (SAMGs) to manage DBAs and BDBAs, including DECs and severe accidents, in accordance with CNSC REGDOC-2.5.2 (Reference 19.6-8) and CNSC REGDOC-2.3.2 (Reference 19.6-13), is noted in Chapter 13, Subsection 13.4.3 and Chapter 18, Subsection 18.3.6.

As noted in preceding discussion, the DNNP senior authorized person on shift is responsible for classifying the level of on-site response and categorizing the event which determines the level of off-site response per the PNERP (Reference 19.6-4).

Off-site emergency notifications are made following categorization. The time requirement for the plant to notify the Provincial contact point is within 15 minutes after the event has been categorized. This is the Provincially set target for notification performance, and a regulatory requirement as noted in CNSC REGDOC-2.10.1 (Reference 19.6-2). The information identified in the official notification form should be confirmed as part of the notification process. The designated regional contact points also receive the same emergency notification shortly after the province. Regulatory notification to the CNSC is made after the off-site agency and ERO notifications have been completed. CNSC notification target time is within 15 minutes of ERO activation.

The communication systems provide intra-plant and plant-to-offsite communications during normal operation, transients, fire, accidents, off-normal phenomena, and security related events are described in Chapter 9A, Subsection 9A.9.1.

OPG implements radiological protective measures to As Low As Reasonably Achievable (ALARA). During the control of an emergency, the applicable emergency dose limits shall be treated as discrete and separate from the normal occupational dose limits prescribed in N-PROC-RA-0019, "Dose Limits and Exposure Control" (Reference 19.6-9).

The design provides safety features for ensuring that doses to operating personnel at the nuclear power plant will be maintained below the dose limits and will be kept ALARA, and that the relevant dose constraints are taken into consideration. The design features for radiation protection are described in Chapter 12, Section 12.3.

NEDO-33969 REVISION 0  
NON-PROPRIETARY INFORMATION

During nuclear emergency events, OPG has a responsibility to communicate with the public, media, stakeholders, and employees. N-STD-AS-0010, "Nuclear Crisis Communications Standard" (Reference 19.6-10) governs public communications on behalf of OPG Nuclear in the event of a nuclear emergency and ensures consistent and coordinated public information response. OPG also supports the province and municipalities who provide coordinated communications under the jurisdiction of the PNERP (Reference 19.6-4). A Joint Information Centre, coordinated by the Provincial Chief Information Officer, may be established with support from OPG and other stakeholders.

Information related to nuclear emergency response is collected for permanent storage once the event has been terminated.

## 19.2 Emergency Response Facilities

The BWRX-300 ERFs including backups meet the requirements of CNSC REGDOC-2.10.1 (Reference 19.6-2) with the intent of remaining consistent with existing OPG Nuclear ERFs:

1. EOC located on-site and referred to as the ERF in CNSC REGDOC-2.5.2 (Reference 19.6-8). The EOC is located near the MCR, where site shift staff assemble to manage and coordinate event response.
2. Site Management Centre located on-site, but outside of the protected area. It is referred to as the Technical Support Centre in CNSC REGDOC-2.5.2 (Reference 19.6-8). The Site Management Centre is staffed with on-call teams of ERO personnel, which includes technical support.

Note that CNSC REGDOC-2.5.2 defined the emergency support facilities as consisting of an on-site ERF (item 1) and a Technical Support Centre (item 2).

3. An ERF located off-site and outside of the plume exposure planning zone. As the design progresses, further details will be included in the Pre-Operational Safety Analysis Report.

The EOC and Site Management Centre contain the equipment necessary to gather, store and display data needed in these facilities to analyze plant conditions. This equipment serves the same function as a Safety Parameter Display System (SPDS), described in Chapter 7, Section 7.3, similar to that in the MCR. Information on radiological conditions in the plant and its immediate surroundings, and about meteorological conditions in the vicinity of the plant, are accessible from the EOC and Site Management Centre. The plant information necessary to support overall emergency response from the MCR, SCR, EOC, Site Management Centre and ERF located off-site and outside the plume exposure planning zone is provided either directly by the Unit Data Highway or via the Plant Data Highway unidirectional boundary device interface, depending on security and Instrumentation and Control requirements related to each location. Further details of instrumentation and controls are described in Chapter 7.

The EOC and Site Management Centre includes secure means of communication with the MCR, SCR and other important points in the plant, and with on-site and off-site EROs per CNSC REGDOC-2.5.2 (Reference 19.6-8). See Chapter 9A, Subsection 9A.9.1 for further details of the communication systems.

The BWRX-300 design includes an MCR, as well as an SCR that is physically and electrically separate from the MCR. The concept of having both the MCR and SCR is no single event can affect both.

The BWRX-300 MCR, located in the Control Building (CB), is the primary location for plant monitoring and control.

In the current concept design, the SCR is located in the Reactor Building. A qualified route exists from the MCR to the SCR for events which necessitate evacuation of the MCR. An alternate route exists for fire events which does not go through the same fire zones as the primary route (refer to Chapter 6, Section 6.4).

The instrumentation and controls in the MCR and SCR are described in Chapter 7, Sections 7.5 and 7.6, respectively.

### **19.3 Capability of the Operating Organization for the Assessment of Potential Radioactive Releases in Accident Conditions**

Accident assessment techniques are employed to determine the extent of on-site radiation impact and to predict the off-site radiation consequence to the public. Processes and methodology include determination of system status from plant parameters, radiological source term measurements, core or fuel damage assessment techniques, meteorological information, off-site dose projections and dose assessment verification using off-site survey results.

The Automated Near Boundary Gamma Monitoring System is located around the DNGS nuclear facility perimeter. This fixed radiological detection and monitoring system may be utilized for BWRX-300 in order to meet CNSC REGDOC-2.10.1 (Reference 19.6-2) requirement to have real-time fixed radiological detection and monitoring capabilities around the nuclear facility perimeter with appropriate backup power. Use of DNGS monitoring system will be assessed as part of BWRX-300 detailed design. In the event of an accident release of radiation, the monitoring system is capable of providing data needed to characterize radiological conditions around the plant and assess the need for off-site action – see Chapter 12, Subsection 12.5.1. During the response phase, technical data, including survey results, shall be made available to the Provincial EOC and CNSC.

In compliance with CNSC REGDOC-2.5.2 (Reference 19.6-8), Instrumentation and Control equipment are provided such that essential information is available to support plant procedures during normal operations and following DBAs and DEC's.

Refinement and selection of designated instruments and displays for accident monitoring and control are based on those deemed necessary to provide the parameters required to accomplish the goals of EOPs and SAMGs. Functional requirement analysis, allocation of functions, and task analysis identify an independent list of the necessary parameters using human factors principles. The preliminary list of accident monitoring variables is provided in Chapter 7, Table 7.3-2.

In compliance with CNSC REGDOC-2.10.1 (Reference 19.6-2), sufficient Personal Protective Equipment (PPE) and provisions to respond to emergencies and protect the emergency responders for the first 72 hours without off-site assistance are provided. The PPE and response equipment are maintained, calibrated (e.g., electronic dosimeters, radiation instrumentation) and staged for use in an emergency. Habitability design requirements in compliance with CNSC REGDOC-2.5.2 (Reference 19.6-8) for the EOC and Site Management Centre will be included in the Pre-Operational Safety Analysis Report.

#### **19.4 Emergency Preparedness for Multiple Unit Sites**

As noted in Section 19.1, on-site emergency arrangements will be submitted with the Licence to Operate application under Commitment D-O-2 (Reference 19.6-5), to ensure emergency preparedness and response is in place prior to fuel-in commissioning.

Additional staffing required for the BWRX-300 specific ERO will be determined. The ERO staffing requirements will be developed through systematic analysis, as described in Chapter 18, Subsection 18.2.4.

Off-site emergency service agreements are described in Chapter 2, Subsection 2.11.5, Administrative Measures with External Organizations. OPG continues to collaborate with the Province of Ontario and other external organizations responsible for off-site nuclear emergency planning to ensure the implementation of their respective emergency plans and related protective actions will accommodate the lifecycle of BWRX-300.

Mutual aid agreements exist in current staffing plans and response procedures. The non-incident station or departments and corporate office may provide spokespeople and technical staff for communication and media centre support.

By virtue of the authority and mandate, the off-site ERF is able to draw on the collective aid of all OPG Nuclear staff and resources for emergency response purposes. This also extends into the entire resource pool of OPG to support an incident at a nuclear generating site.

As stated in CNSC REGDOC-2.10.1 (Reference 19.6-2), an effective response to an emergency requires strong linkages between accident management, which is covered in CNSC REGDOC-2.3.2 (Reference 19.6-13) and emergency preparedness. Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents are part of Defense Line 4 (DL4) of the defence-in-depth approach (refer to Chapter 3, Subsection 3.1.6). The fourth DL is split into DL4a and DL4b, providing independent provisions for prevention and mitigation of DEC. DL4a is for DEC that occur without core damage, and DEC progressing to core damage are DL4b. The use of non-permanent equipment, known as Emergency Mitigating Equipment (EME), is an acceptable DL4b provision to restore lost safety functions, but not to be the regular means to achieve these functions in accident conditions. In compliance with CNSC REGDOC-2.3.2 (Reference 19.6-13), BWRX-300 will have EME, and its use is covered in EMEGs. EME will continue to be assessed as part of BWRX-300 design.

### **19.5 Emergency Preparedness for Internal Fire Hazards**

A comprehensive fire protection program covers control of combustible materials, control of ignition sources, as well as fire response strategies, to meet the requirement of CSA N293S1, "Supplement No. 1 to N293-12, Fire protection for nuclear power plants (application to small modular reactors)" (Reference 19.6-12). As stated in Chapter 3, Subsection 3.4.1, this program will provide additional layers of defence in addition to the fire protection features in place against internal fire hazards. The program aims to prevent, mitigate, and manage emergencies that arise from internal fire hazards that may lead to radiological and nuclear emergencies. Refer to Chapter 9A, Section 9A.6 for the description of the Fire Protection System.

## **19.6 References**

- 19.6-1 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan (CNEP)," Ontario Power Generation.
- 19.6-2 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 19.6-3 CSA N1600, "General requirements for nuclear emergency management programs," CSA Group.
- 19.6-4 "Provincial Nuclear Emergency Response Plan (PNERP): Master Plan," Executive Council of Ontario.
- 19.6-5 NK054-PLAN-01210-00002, "DNNP Nuclear Emergency Preparedness Plan," Ontario Power Generation.
- 19.6-6 CNSC Regulatory Document REGDOC-2.3.1, "Conduct of Licensed Activities: Construction and Commissioning Programs."
- 19.6-7 IAEA Safety Standards Series NO. GS-G-2.1, "Arrangements for Preparedness for a Nuclear or Radiological Emergency," International Atomic Energy Agency.
- 19.6-8 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 19.6-9 N-PROC-RA-0019, "Dose Limits and Exposure Control," Ontario Power Generation.
- 19.6-10 N-STD-AS-0010, "Nuclear Crisis Communications Standard," Ontario Power Generation.
- 19.6-11 D-INS-03490-10009, "Shift Manager," Ontario Power Generation.
- 19.6-12 CSA N293S1, "Supplement No. 1 to N293-12, Fire protection for nuclear power plants (application to small modular reactors)," CSA Group.
- 19.6-13 CNSC Regulatory Document REGDOC-2.3.2, "Operating Performance – Accident Management."
- 19.6-14 D-INS-03490-10007, "Response to Warning Signal," Ontario Power Generation.
- 19.6-15 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33970

Revision 1

October 7, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 20  
Environmental Aspects**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release
1	Section 20.4 Section 20.6 Section 20.7 Section 20.8.2	Incorporate corrections per customer acceptance review.

### ACRONYM LIST

Acronym	Explanation
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
DBA	Design Basis Accident
DEC	Design Extension Condition
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EA	Environmental Assessment
EFS	Equipment and Floor Drain System
EIS	Environmental Impact Statement
EMP	Environmental Monitoring Program
ERA	Environmental Risk Assessment
GEH	GE Hitachi Nuclear Energy
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating, Ventilation, and Cooling System
ISO	International Organization for Standardization
LWM	Liquid Waste Management System
NRC	U.S. Nuclear Regulatory Commission
OGS	Offgas System
OPG	Ontario Power Generation
PREMS	Process Radiation and Environmental Monitoring System
PSAR	Preliminary Safety Analysis Report
SI	International System of Units
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components

## TABLE OF CONTENTS

<b>20.0 ENVIRONMENTAL ASPECTS .....</b>	<b>20-1</b>
20.1 Introduction .....	20-1
20.2 General Aspects of the Environmental Assessment .....	20-1
20.2.1 References.....	20-1
20.3 Environmental Principles and Regulations.....	20-2
20.4 Environmental Management System .....	20-2
20.4.1 References.....	20-2
20.5 Site Characteristics Important to Environmental Impact .....	20-4
20.5.1 References.....	20-5
20.6 Best Available Technology and Techniques Economically Available Design Features .....	20-6
20.7 Environmental Impact of Construction .....	20-7
20.7.1 References.....	20-8
20.8 Environmental Impact of Normal Operation .....	20-10
20.8.1 Direct Radiation from Buildings and Facilities.....	20-10
20.8.2 Radiological Airborne and Liquid Effluent Releases .....	20-10
20.8.3 Control of Radioactive Discharges to the Environment.....	20-12
20.8.4 Non-Radioactive Contaminants .....	20-12
20.8.5 Authorized Limits and Operational Targets for Discharges and Releases .....	20-13
20.8.6 Radiological Impacts of Normal and Abnormal Operation .....	20-13
20.8.7 Measures and Controls to Limit Adverse Impacts During Operation ....	20-13
20.8.8 References.....	20-14
20.9 Environmental Impact of Postulated Accidents Involving Radioactive Releases	20-15
20.9.1 Design Basis Accidents .....	20-15
20.9.2 Design Extension Conditions .....	20-15
20.9.3 Measures and Controls to Limit Adverse Impacts During Accidents ....	20-15
20.9.4 Emergency Actions in Response to External Events .....	20-15
20.9.5 References.....	20-15
20.10 Environmental Impact of Facility Decommissioning .....	20-16
20.10.1 Reference .....	20-18
20.11 Environmental Measurements and Monitoring Programs .....	20-19

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

20.11.1	Darlington New Nuclear Plant Environmental Assessment Follow-up Program .....	20-19
20.11.2	Environmental Monitoring Program .....	20-20
20.11.3	Effluent Monitoring Program .....	20-21
20.11.4	Groundwater Monitoring Program.....	20-21
20.11.5	Monitoring Systems and Alarms .....	20-22
20.11.6	References.....	20-22
20.12	Records of Radioactive Releases and Availability of Information to the Authorities and the Public .....	20-24
20.12.1	Record Storage and Retention .....	20-24
20.12.2	Data Availability to External Bodies .....	20-25
20.12.3	References.....	20-26
20.13	Appendix A: Annual Average Gaseous Effluent Releases for BWRX-300 .....	20-28
20.13.1	Methodology .....	20-28
20.13.2	Results .....	20-28
20.13.3	References.....	20-36
20.14	Appendix B: BWRX-300 Annual Average Liquid Effluent Activity Releases for BWRX-300 .....	20-37
20.14.1	Methodology .....	20-37
20.14.2	Results .....	20-37
20.14.3	References.....	20-40

**LIST OF TABLES**

Table 20.7-1: Maximum Emission Rates During Site Preparation and Construction Phase...	20-9
Table 20.8-1: Maximum Emission Rates from Fuel Combustion During Operations and Maintenance Phase .....	20-14
Table 20.13-1: BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr) .....	20-29
Table 20.13-2: BWRX-300 Annual Airborne Releases and Concentrations Including AOOs .....	20-33
Table 20.14-1: Total Annual Average Liquid Effluent Release Including AOOs .....	20-38

## **LIST OF FIGURES**

None.

## **20.0 ENVIRONMENTAL ASPECTS**

### **20.1 Introduction**

This Chapter discusses the environmental aspects related to the construction, operation and decommissioning of the Boiling Water Reactor, 10th Design – 300 MWe (BWRX-300) Small Modular Reactor (SMR).

A brief overview of OPG's Environmental Management System, site characteristics including relevant meteorological data and environmental assessment follow-up program are included. Environmental impacts of construction, normal operation, accidents, and decommissioning are summarized.

### **20.2 General Aspects of the Environmental Assessment**

The Darlington New Nuclear Project (DNNP) Environmental Assessment (EA), including the Environmental Impact Statement (EIS) (Reference 20.2-1), was submitted in 2009 and accepted by the Government of Canada in 2012. The Environmental Assessment assessed the impact of the preparation of the site; construction of up to four new reactors with a maximum generation capacity of 4800 MW(e), and associated facilities; the operation and maintenance of the reactors and related facilities for approximately 60 years, the management of conventional and radioactive waste; and the decommissioning of the nuclear reactors and associated facilities. The EIS concluded DNNP does not have significant adverse environmental effects if the mitigation measures proposed are implemented.

A number of reactor technologies were considered in the EA. For assessment purposes, a Plant Parameter Envelope was used to provide a bounding framework of plant design and site parameter values from the multiple reactor technologies.

The DNNP – Site Preparation Licence Renewal Activity Report – Environment (NK054-REP-01210-00110) (Reference 20.2-2) concludes that any minor changes in baseline conditions since the time the EA was conducted are adequately addressed by existing commitments and do not alter the residual adverse effects of the project.

The current Nuclear Power Reactor Site Preparation Licence 18.00/2031 for the DNNP was renewed in 2021 and expires on October 11, 2031 (Reference 20.2-3).

The BWRX-300 SMR was selected for DNNP in 2021. A comprehensive review of the EIS confirms that results of the EIS remain valid for deployment of the BWRX-300. The EIS review is submitted to the CNSC to support the Licence to Construct application.

#### **20.2.1 References**

- 20.2-1 NK054-REP-07730-00029, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 20.2-2 NK054-REP-01210-00110, "DNNP – Site Preparation Licence Renewal Activity Report – Environment," Ontario Power Generation.
- 20.2-3 PRSL 18.00/2031, "Power Reactor Site Preparation Licence," Ontario Power Generation



### **20.3 Environmental Principles and Regulations**

The construction, operation, and decommissioning of the BWRX-300 at OPG complies with the regulations and guidance for protecting the environment at the DNNP site listed in Chapter 1, Appendix B.

### **20.4 Environmental Management System**

As outlined in OPG's Environmental Policy (Reference 20.4-1), OPG shall meet compliance obligations, including any environmental commitments that it makes, with the objective of exceeding these compliance obligations, where it makes business sense. The Environmental Policy applies to all OPG, contractor, and subcontractor employees working on DNNP activities.

Ontario Power Generation's Environment Health and Safety Managed Systems program (Reference 20.4-2) governs the design and execution of OPG's Environmental Management System in accordance with OPG-POL-0021, Environmental Policy (Reference 20.4-1). The scope of OPG's Environmental Management System includes the management of activities related to design, supply chain, construction, commissioning, operation (including emissions, effluent, waste management, and other support processes), and decommissioning of electricity generating stations, and associated properties, auxiliary buildings, and structures solely owned and/or operated by OPG, or sites operated by OPG on behalf of OPG's partner owners.

The Environmental Management System is consistent with the CNSC Regulatory Document REGDOC-2.9.1 (Reference 20.4-3). Additionally, this program is registered to and compliant with the International Organization for Standardization (ISO) 14001 Environmental Management Systems standard (Reference 20.4-4).

All phases of the DNNP will be executed in a manner that conforms to the requirements of OPG's Environmental Policy and Environmental Management. In the case where OPG is an owner in an integrated project delivery contract, it is OPG's responsibility to ensure ongoing oversight through its management system. The other contract parties will define and implement their own management system compliant with applicable current standards or implement an alternative management system as agreed to by the project partners. Ontario Power Generation provides other contract parties with the DNNP Environmental Management and Protection Plan (Reference 20.4-5), which defines OPG's expectations to ensure protection of the environment.

The ISO 14001 standard (Reference 20.4-4) requires that any person performing work or services on an OPG site, where such work or service has the potential to cause a significant environmental impact, should be aware of the requirements and importance of the Environmental Management System and be competent to perform the work assigned. Any contractor who performs an operation or activity that has the potential for significant environmental impacts shall implement measures and controls to minimize the environmental risk (e.g., documented procedure controls).

#### **20.4.1 References**

- 20.4-1 OPG-POL-0021, "Environmental Policy," Ontario Power Generation.
- 20.4-2 OPG-PROG-0005, "Environment Health and Safety Managed Systems," Ontario Power Generation.
- 20.4-3 CNSC Regulatory Document REGDOC-2.9.1, "Environmental Protection: Environmental Principles, Assessments and Protection Measures."

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

- 20.4-4 ISO 14001, Environmental Management Systems," International Organization for Standardization.
- 20.4-5 NK054-PLAN-07730-00022, "Environmental Management and Protection Plan for Site Preparation," Ontario Power Generation.

## **20.5 Site Characteristics Important to Environmental Impact**

The site characteristics that are important in terms of environmental impact are addressed in detail in Chapter 2, Section 2.8. The information is summarized below.

Relevant meteorological characteristics include temperature, precipitation, wind speed and wind direction:

1. Temperature: From 2016-2019, the highest (21.5°C) and the lowest (-4.1°C) 4-year average monthly temperatures at the Darlington Nuclear site occurred in the months of July and January, respectively. The mean daily maximum temperature (23.0°C) was recorded in August 2016, and the mean daily minimum temperature (-6.4°C) was recorded in January 2019 (Reference 20.5-1).
2. Precipitation: The 1981-2010 climate normal data from the Bowmanville Mostert station reported an average annual precipitation of approximately 866 mm. Snowfall represented less than 11% of total precipitation. Total monthly precipitation averages range from approximately 50.5 mm in February to approximately 98.7 mm in September (Reference 20.5-1).
3. Wind Speed: From 2013-2019, the average wind speed measured at the on-site meteorological towers was approximately 2.4 m/s; calm winds of less than 2 m/s were reported ~37% of the time. The prevailing winds for these years were also measured to be from the North-West sector; North (9.6% of the time), followed by the West direction (8.9% of the time) (Reference 20.5-1).
4. Wind Direction: In 2021, the landward sector at the Darlington Nuclear site that wind predominantly blew towards, was the East-North-East sector (wind from West-South-West), based on the average annual wind frequencies at a 10 m height. Over all sectors, the wind predominantly blew from the North and West sectors; the dominant wind direction was North-North-West (10.38% of the time), followed by West (9.98%) and North-West (9.82%) (Reference 20.5-2).

Relevant hydrological characteristics include stormwater runoff, municipal water supply sources, lake currents and temperature, and water intake and return structures. In the developed parts of Darlington Nuclear, stormwater is collected in natural channels/swales and constructed outfalls and conveyed off-site to Lake Ontario. The lake current in the nearshore region is overall easterly; for the 2012-2016 period, speeds typically ranged from about 9-18 cm/s. Lake-wide surface temperatures typically range from freezing in the winter to approximately 20°C in the summer. The water temperatures recorded from December 2011 to March 2012 and from December 2011 to April 2012 in the Darlington Nuclear study area had an average temperature of 3.8°C and 4.4°C, respectively. Water is drawn from the lake by Darlington Nuclear Generating Station (DNFS) via an 80-m diameter, porous bottom, submerged intake structure located at a distance of 700 m from the shoreline at a depth of approximately 10 m. Cooling water is returned to Lake Ontario from DNFS via a 900-m long submerged diffuser extending approximately 1,600 m from the shoreline (Reference 20.5-1).

Relevant hydrogeological characteristics include aquifer type and groundwater flow. Groundwater aquifers at the Darlington Nuclear site has been categorized into three hydrostratigraphic units: Shallow/Water Table, Interglacial Deposits, and Shallow Bedrock. Within the Shallow/Water Table hydrostratigraphic units, groundwater flows from north to south approaching Lake Ontario. General flow patterns within the interglacial deposits hydrostratigraphic units and shallow bedrock hydrostratigraphic units are similar to the shallow/water table hydrostratigraphic units (Reference 20.5-3).

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

The dominant ecological feature of the Darlington Nuclear site is meadow (24%), followed by thicket (14%), woodland (5%), and swamp (5%). In general, the Darlington Nuclear site has four main areas: in the northwest there are sports fields, a large settling pond (Coot's Pond), and Bobolink Hill comprised of cultural meadow and cultural thicket; in the north-east there are agricultural fields, cultural thicket, and deciduous forest as well as three constructed wetland ponds (Treefrog, Dragonfly and Polliwog ponds); the south-east is mostly cultural meadow; and in the south center and south-east is the DNGS (Reference 20.5-2).

The census data for the region used in the most recent Darlington Nuclear Site-Specific Survey are for 2016. A population of approximately 500,000 resides within a 30 km radius of the Darlington Nuclear site, based on 2016 census data. The bulk of this population (approximately 88% or 478,634 individuals) resides West of the Darlington Nuclear site, in the West-South-West to North-North-West sectors, while approximately 12% (64,575 individuals) reside East of the Darlington Nuclear site in the North to East North-East sectors. The majority of residents in Durham Region live in urban areas. Over 90% of the population in Pickering, Ajax, Oshawa, and Whitby reside in urban areas, whereas, the townships of Brock, Scugog and Uxbridge represent the greatest percentage of the rural population in Durham. Children under the age of 15 comprised 18.0% of Durham's population in 2016, while young persons (aged 15-29), adults (aged 30-64) and older adults (aged 65+) comprised 19.2%, 49.4% and 14.4%, respectively (Reference 20.5-1).

#### **20.5.1 References**

- 20.5-1 D-REP-07701-00001 R001, "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 20.5-2 N-REP-03443-10027 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 20.5-3 NK054-REP-01210-0001 R000, "Darlington New Nuclear Project Supporting Environment Studies – Environment," Ontario Power Generation.

## **20.6 Best Available Technology and Techniques Economically Available Design Features**

BWRX-300 design features that reduce effluent emissions, which may be considered Best Available Technology and Techniques Economically Available include:

- Material selection – keeps the concentration of radiation in the coolant low
- Fuel reliability - programs and procedures that govern operational activities associated with BWRX-300 core management regarding fuel reliability are based on guidelines established by GEH utilizing decades of experience with Global Nuclear Fuel
- Gaseous radwaste management - the Offgas system (OGS) is a state-of-the-art feature that holds up the release of noble gases enabling them decay to low levels before being released to the environment
- Liquid Waste Management System (LWM), and minimal release philosophy – state-of-the-art liquid waste management design that minimizes liquid releases
- Leak detection systems – prevents and mitigates leaks that could reach the environment
- Reactor coolant cleanup systems – the Condensate Filters and Demineralizers system is a state-of-the-art design that capture contamination and lowers effluent releases
- Heating Ventilation and Air Conditioning (HVAC), High Efficiency Particulate Air (HEPA) and charcoal filtration systems – filtering of HVAC discharge to the environment
- The Process Radiation and Environmental Monitoring System (PREMS) provides continuous and periodic monitoring to allow determination of the content of radioactive material in various gaseous, liquid process, and effluent streams

The Best Available Technology and Techniques Economically Available principle was applied via an assessment completed in 2013 to meet D-C-1.1, which compared Once Through Cooling and Mechanical Draft Cooling Towers. Once Through Cooling was recognized as the better performing option with respect to the following evaluation areas: visual effect, local climate change, terrestrial habitat, and excavated materials. The difference between the two options for impacts on fish (impingement, entrainment, and thermal plume) was minimal and did not compensate for the better environmental performance of the Once Through Cooling system. Following the decision to consider and SMR for the DNNP site, a review of the Best Available Technology and Techniques Economically Available work was completed in 2021 to confirm that the conclusions of the original Best Available Technology and Techniques Economically Available work were unchanged and that D-C-1.1 could remain closed.

## **20.7 Environmental Impact of Construction**

The DNNP is not expected to contribute radiological releases during the construction stage. Potential sources of radiation during construction include adjacent nuclear facilities and sealed radioactive sources used during the plant construction. Adjacent nuclear stations to DNNP land include DNGS, as well as the Tritium Removal Facility and Darlington Waste Management Facility immediately to the West. Pickering Nuclear Generating Station is approximately 30 km West-South-West. The environmental impacts of these stations on the surrounding environment are well characterized in Environmental Monitoring Program reports and Environmental Risk Assessments (ERA). The results of the 2021 Environmental Monitoring Program relating to Darlington Nuclear are summarized in Chapter 2, Subsection 2.9.1. The DNNP is not being constructed within the existing operating Darlington Nuclear site facilities, so construction activities will not impact the operation of DNGS, Darlington Waste Management Facility, or the Tritium Removal Facility.

The only anticipated potential radiological impacts from the construction phase may be through disruption and excavation of contaminated soil due to the proximity of DNGS, and potential redirection of the groundwater flow during excavation activities. To determine an updated soil quality baseline, a soil sampling program was conducted in 2019 and a soil characterization program was conducted in 2021 at the DNNP site. In the 2019 program, both conventional and radiological parameters were measured. Screening against appropriate benchmarks indicated that soils were found to be of good quality with the exception of one area identified as the yard waste and building materials storage area where the following parameters exceeded the conventional benchmarks (Reference 20.7-1):

- Cyanide
- Arsenic
- Cadmium
- Chromium
- Cobalt
- Copper
- Lead
- Molybdenum
- Nickel
- Selenium
- Tin
- Zinc
- Petroleum hydrocarbons

In the 2021 soil characterization program, only conventional parameters were measured as assessment showed there are no potential concerns for radiological contaminants. Exceedances in shallow soil (copper, zinc, arsenic, petroleum hydrocarbon F1, petroleum hydrocarbon F2) were identified at seven sampling locations in the DNNP Lands South, which includes two parking lot areas (petroleum hydrocarbon exceedances). In general, the soil exceedances identified above were only marginally above the Ontario Ministry of the Environment, Conservation and Parks

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Table 3 Standards for soil, and would be considered a non-hazardous waste; however, these soils are potentially suitable for reuse at the DNNP site (Reference 20.7-2).

The Environmental Management and Protection Plan (Reference 20.7-3) outlines requirements for soil management and mitigation measures to be followed, such as soil handling, sampling, and disposal procedures, as well as dust control. Therefore, it is not anticipated that there will be impacts from radiological or hazardous material during excavation as part of DNNP construction.

Groundwater flow on the DNNP site may be altered during excavation activities, redistributing any existing radiation in the groundwater. However, groundwater monitoring results at DNNP site for tritium were non-detectable from 2016-2020 (References 20.7-1, 20.7-4 and 20.7-5). Groundwater flow is expected to return to near pre-development conditions within a couple of years after construction is completed.

Construction activities are discussed at a management systems level in Chapter 14, Section 14.2. Field sources such as those used in portable gauges and radiography devices may be used during construction activities and will be operated under separate CNSC issued licences.

During the construction phase of DNNP, management and monitoring of releases from DNNP construction activities is addressed by the DNNP Environmental Management and Protection Plan and Environmental Assessment follow-up monitoring program as well as compliance with the relevant environmental permits and Environmental Compliance Approvals.

Maximum emission rates of environmental pollutants during site preparation and construction, based on industry data for emissions are estimated in Table 20.7-1. Estimates are for activities such as, material handling, grading, and dozing, rock blasting, crush and screen, worker parking lot, parking lot tailpipe, unpaved roads, haul truck tailpipe, paved roads, paved road tailpipe, stationary equipment tailpipe, non-road tailpipe and concrete batching. Tailpipe emissions are from industrial machinery, haul trucks, and worker vehicles driving on the paved parking lot, roads, or on the unpaved roads. The non-road tailpipe activities, such as the “worker parking lot” or “unpaved roads”, are the dust emissions that arise from driving, based on assumed silt content and other parameters.

#### **20.7.1 References**

- 20.7-1 NK054-REP-01210-0001 R000, “Darlington New Nuclear Project Supporting Environment Studies – Environment,” Ontario Power Generation.
- 20.7-2 NK054-REP-07730-00053, “Soil Characterization Report,” Ontario Power Generation.
- 20.7-3 NK054-PLAN-07730-00022, “Environmental Management and Protection Plan for Site Preparation,” Ontario Power Generation.
- 20.7-4 NK38-REP-10140-10030, “2019 Darlington Nuclear Groundwater Monitoring Program Results,” Ontario Power Generation.
- 20.7-5 NK38-REP-10140-10031 R001, “Darlington Nuclear Groundwater Monitoring Program Results,” Ontario Power Generation.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 20.7-1: Maximum Emission Rates During Site Preparation and Construction Phase**

Activity	Maximum Emission Rate (g/s)						
	SPM	PM <sub>10</sub>	PM <sub>2.5</sub>	NO <sub>x</sub>	SO <sub>2</sub>	CO	Acrolein
Material Handling	0.278	0.132	0.020				
Grading and Dozing	1.163	0.267	0.122				
Rock Blasting				0.911	0.114	3.872	
Crush and Screen <sup>1</sup>	-	-	-				
Worker Parking Lot	2.117	0.718	0.071				
Parking Lot Tailpipe	0.00296	0.00296	0.00194	0.11881	0.00189	1.17120	0.00004
Unpaved Roads	1.68	0.48	0.05				
Haul Truck Tailpipe	0.004	0.004	0.003	0.161	0.004	0.042	0.00003
Paved Roads	0.106	0.020	0.003				
Paved Road Tailpipe	0.011	0.011	0.008	0.458	0.008	3.920	0.000
Stationary Equipment Tailpipe	0.166	0.166	0.166	2.344	0.155	0.505	0.00699
Non-Road Tailpipe	0.3249	0.3249	0.3249	7.0581	1.8997	2.7778	0.05064
Concrete Batching	0.1969	0.0702	0.0585				

(1) Crushing and screening will not be conducted on site.



## **20.8 Environmental Impact of Normal Operation**

Liquid and gaseous discharges to the environment, due to normal operational activities will be monitored for conformance with OPG's Environmental Health and Safety Managed Systems (Reference 20.8-1).

### **20.8.1 Direct Radiation from Buildings and Facilities**

Sources of radiological emissions to the environment during operation of the BWRX-300 SMR include radiation from buildings and facilities.

- Radiation shine from buildings and facilities are shielded. The buildings containing radioactive materials are discussed in Chapter 12, Section 12.2. These include:
  - Containment Radiation Sources: Reactor Core, Control Rod Drive System and Reactor Startup Source
  - Reactor Building and Plant Services Area Radiation Sources: Fuel Pool Cooling and Cleanup System and Fuel Pool
  - Turbine Building Source Terms: Reactor Water Cleanup System Radiation Sources, Main Steam and Feedwater Lines, Normal Operating Sources
  - Radwaste Building Radiation Sources: Normal Operating Sources represent sources for off-site release which are low-consequence, low-frequency events
  - Plant Services Area Sources: Contains the Reactor Auxiliary Bay where new and spent fuel is moved in and out of the reactor building

### **20.8.2 Radiological Airborne and Liquid Effluent Releases**

This subsection deals with the sources and parameters required to evaluate radiological airborne and liquid releases during normal plant operation. During operation, effluent releases must be below release limits, which are calculated based on CSA-N288.1 (Reference 20.8-2) to meet dose limits for effluent release cited in SOR/2000-203 (Reference 20.8-3).

The public dose from gaseous and liquid effluent releases during normal operation is expected to be well below regulatory limits.

Annual average gaseous and liquid effluent releases for long-term normal operation of BWRX-300, including Anticipated Operational Occurrences (AOOs), are included in Appendices A and B, respectively. The annual gaseous and liquid effluent releases are the basis for determination of dose consequences to the public from routine releases of airborne and liquid effluents from the BWRX-300 DNNP plant. For the purposes of this PSAR, the gaseous and liquid effluent releases including AOOs are used as conservative inputs for the estimate of normal operational dose from the BWRX-300 at the DNNP site.

The gaseous and liquid effluent release analyses are performed with the GALE-BWR methodology in U.S. Nuclear Regulatory Commission (NRC) NUREG-0016, Revision 2 (Reference 20.8-4). GALE-BWR calculates expected releases based on 1) standardized coolant activities derived from BWR operating data, 2) release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, 3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environs, and 4) information received on the operation of nuclear power plants.

Description, validation, and extent of application of the GALE-BWR 3.2 code (Reference 20.8-4) is provided in Chapter 3, Appendix 3F.

#### **20.8.2.1 Radiological Airborne Releases**

BWRX-300 annual airborne effluent release activities including AOOs are reported in Table 20.13-2. The annual airborne effluent release activities are calculated using the normal operation coolant concentrations provided in Chapter 11, Section 11.1, and the calculation is described in detail in Appendix A. The BWRX-300 plant releases airborne effluents through the plant stack.

#### **20.8.2.2 Radiological Liquid Releases**

The LWM described in Chapter 11, Section 11.2 is designed to monitor and process all radioactive liquid streams and provide wastewater management for those streams. Under normal operation conditions, water management does not result in routine release of radioactive effluents. Under some conditions where high wastewater inventory is processed (refueling or abnormal conditions), some processed radioactive liquid effluents may be released. By administrative control, the discharge of these effluents through the discharge line is adjusted so that the discharge meets the regulatory limits.

BWRX-300 annual average liquid effluent release activities including AOOs are reported in Table 20.14-1. The annual average liquid effluent release activities are calculated using the normal operation coolant concentrations provided in Chapter 11, Section 11.1, and the calculation is described in detail in Appendix B.

The BWRX-300 LWM implements a minimal release plan for the treatment of liquid radwaste. Minimal release is a philosophy of processing wastewater such that it is sent back into the system and released into local bodies of water. Thus, the release of liquid effluents determined in this analysis is assumed for conservatism. If a liquid discharge from the LWM were to occur, it would be diluted and discharged in the plant cooling water discharge stream.

#### **20.8.2.3 Airborne and Liquid Effluent Doses for Members of the Public**

Doses to members of the public are modeled with the IMPACT code in accordance with the CANDU Owners Group Derived Release Limit Guidance Document (Reference 20.8-5) and CSA-N288.1 (Reference 20.8-2). For the purposes of this PSAR, the annual release of airborne and liquid effluents including AOOs from Tables 20.13-2 and 20.14-1, were used as conservative inputs for the estimate of dose during normal operations to off-site members of the public, known as potential critical groups. The potential critical groups considered in the dose model align with those identified in the Environmental Monitoring Program (Reference 20.8-6).

The critical group dose from the operation of one BWRX-300 at the DNNP site (including AOOs) is conservatively estimated to be 4.87E-04 mSv per year. This is a very small fraction of the regulatory dose limit of 1 mSv per year cited in SOR/2000-203 (Reference 20.8-3).

### **20.8.3 Control of Radioactive Discharges to the Environment**

On-site and off-site process radiation systems and environmental monitoring programs are discussed in Section 20.11. As indicated in Chapter 11, Section 11.5, PREMS monitors the release of airborne and liquid effluents from the station, alarming upon the release of activity concentration above action levels.

As described in Chapter 12, Section 12.3, design features and measures within the plant minimize radiation and contaminants that could lead to discharges to the environment. These include:

1. Sump vents are located near the room exhaust vent register to control airborne radioactivity released from discharges to the sump.
2. All contaminated or potentially contaminated systems shall be isolated from the outside environment or are otherwise equipped with isolation devices to control their discharges.
3. Systems designed to discharge to the environment are equipped with radiation monitoring devices to automatically isolate the discharge if high activity is detected.
4. To prevent radiation overexposure to detect leaks, and to prevent high-activity discharge, the atmospheres of the rooms and stacks are monitored for particulates, noble gases, and iodine.
5. Effluents from radioactive waste tanks of the LWM are discharged to the tank room in the general vicinity of the room's ventilation exhaust register. Tank vents are designed such that liquid and entrained solids do not splash out (e.g., use of baffle plates). Hard connections directly to the ventilation system are avoided to ensure liquid or moisture does not propagate through the ventilation system damaging the HEPA filters.
6. During normal operation purging, the containment exhausts air through the HVAC HEPA filter unit prior to stack discharge.
7. The ventilation system is designed to sweep air from the Spent Fuel Pool surface, thereby removing the major portion of potential airborne contamination.
8. The derivation of trip setpoints for effluent and discharge radiation monitors will be specified in the plant operating procedures. Setpoints will be established on the basis of discharge flow rates and the resulting off-site doses.

Environmental worker qualifications and training is addressed in OPG-PROG-0005 (Reference 20.8-1).

### **20.8.4 Non-Radioactive Contaminants**

Maximum emission rate from fuel combustion contaminants during operations and maintenance phase are estimated in Table 20.8-1. Sources of fuel combustion contaminants include vehicles, standby emergency power generators, fire pump diesel engines and security diesel generator. Estimates in Table 20.8-1 are based on industry data for emissions.

The DNNP on-site chemicals and gases include, nitrogen, hydrogen, diesel fuel, turbine oil, sodium hypochlorite, sodium bisulphite, Captor (R) calcium thiosulphate, gasoline, propylene glycol, tetrafluoroethane (R134a Refrigerant), Noble Metal Solution and depleted zinc oxide. These DNNP on-site gases and chemicals are itemized in Chapter 2, Section 2.4, Table 2.4-1. Depleted zinc oxide is deposited on plant surfaces of reactor vessel components; some depleted zinc oxide is removed by the reactor water cleanup system (Cleanup water) and some is removed in solid waste. Sodium hypochlorite, sodium bisulphite and Captor (R) calcium thiosulphate are released into the lake in compliance with an approved Permit. Propylene Glycol is not released to the environment. Tetrafluoroethane (R134a Refrigerant) is used in the chiller system for the

Radwaste Building, this is a closed system, and therefore tetrafluoroethane is not released to the environment. Noble Metal Solution deposits on the surfaces of reactor vessel components. The zinc and the Noble Metal Solution both deposit on the fuel surfaces and are not released to the environment. The containment system is inerted with nitrogen during different modes and is purged to the environment via the plant ventilation system, while being monitored to ensure there is no radiological release. Hydrogen is released to the environment through the OGS. Diesel fuel and gasoline are used for engines and diesel generators and are protected from accidental release to the environment. Turbine oil is contained in the plant and is protected from accidental release to the environment.

There are no other non-radiological contaminants that would be released as effluents to the air or water.

Appropriate design measures will be taken to control the release of non-radioactive contaminants to the environment. OPGs Spill management program and chemistry control standard will apply for control of non-radioactive contaminants.

#### **20.8.5 Authorized Limits and Operational Targets for Discharges and Releases**

The DNNP will not generate emissions or contribute releases of nuclear or hazardous substances during the construction phase. Ontario Power Generation will develop authorized release limits and action levels for DNNP's radiological releases prior to the operations phase, in accordance with the relevant regulatory documents and standards in place at that time.

Release limits for non-radiological (hazardous) substances will be determined in accordance with relevant Environmental Compliance Approvals and other licences, permits, and regulatory requirements and standards for each phase (site preparation, construction, fuel-out commissioning, operation).

#### **20.8.6 Radiological Impacts of Normal and Abnormal Operation**

The ALARA design requirements, keep radiation exposures ALARA during normal operation or anticipated operational occurrences. Planned radioactive material releases are kept below regulatory limits. Normal operation doses are discussed in Subsection 20.8.2. Abnormal and accident doses are discussed in Section 20.9.

#### **20.8.7 Measures and Controls to Limit Adverse Impacts During Operation**

Design measures to control gaseous and liquid radioactive discharges to the environment during normal operation are described in Chapters 11 and 9A and summarized below.

The OGS, described in Chapter 11, Section 11.3, processes, and controls the release of gaseous radioactive effluents to the site environment, thereby maintaining exposure of persons in unrestricted areas to radioactive gaseous effluents ALARA and within prescribed limits during reactor power operation per CNSC REGDOC-2.5.2, Section 8.11.3 (Reference 20.8-7).

The OGS minimizes the release of radioactive material into the atmosphere by delaying release of the offgas process stream. This delay, using activated charcoal adsorber beds, is sufficient to achieve adequate decay before the process offgas stream is discharged from the plant. The OGS is designed with redundant components to ensure system operation continues in the event of failure of these components and for accessibility for maintenance.

Chapter 9A, Section 9A.5 describes the Station Heating, Ventilation, and Cooling System (HVS) which serves various areas of the power block during normal operation. The HEPA filters from the Exhaust Air Handling Units at each building assist in ensuring radioactive material entrained in gaseous effluent will not exceed the limits specified in SOR 2000-203 (Reference 20.8-3), for normal operations and anticipated operational occurrences. All radiologically controlled areas

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

requiring vent to the Continuous Exhaust Air Plenum which collects, mixes, and dilutes potentially radioactive discharge air from various buildings and releases the mixed air to atmosphere via the Plant Vent Stack. The Plant Vent Stack is monitored by PREMS as discussed in Chapter 11, Subsection 11.5.

The gaseous effluent treatment systems are designed to limit the dose to off-site persons from routine station releases to significantly less than the limits in SOR 2000-203 (Reference 20.8-3).

The LWM, described in Chapter 11, Section 11.2, collects liquid radioactive waste generated throughout the plant as a result of normal operation, separates and filters the waste, then returns the filtered water to the plant systems for reuse. The BWRX-300 is designed such that under normal conditions the need to release liquid effluent to the environment is minimized. In cases where liquid effluent is released, the PREMS system monitors the effluent stream sending a signal to automatically isolate the discharge line if it senses radioactivity above action levels.

Ontario Power Generation's Standard Controlling Radiation Exposure to the Public and the Environment to ALARA establishes the minimum requirements to keep radiological exposures and doses to the public and the environment ALARA, taking social and economic factors into account.

**Table 20.8-1: Maximum Emission Rates from Fuel Combustion During Operations and Maintenance Phase**

Contaminant	24-hour (g/s)	Annual (g/s)
Acrolein	1.11E-04	1.11E-04
Carbon Monoxide	1.26E+01	3.50E+00
Nitrogen Oxides	1.46E+01	6.31E-01
SPM	4.92E-01	2.01E-02
PM10	4.92E-01	2.01E-02
PM2.5	4.53E-01	1.57E-02
Sulphur Dioxide	5.18E-03	5.18E-03

#### 20.8.8 References

- 20.8-1 OPG-PROG-0005, "Environment Health and Safety Managed Systems," Ontario Power Generation.
- 20.8-2 CSA-N288.1-14, "Guidelines for modelling radionuclide environmental transport, fate and exposure associated with normal operations of nuclear facilities," CSA Group.
- 20.8-3 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 20.8-4 USNRC NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR)."
- 20.8-5 COG-06-3090-R2-I, "Derived Release Limits Guidance," CANDU Owners Group Inc.
- 20.8-6 N-REP-03443-10027 "2021 Results of Environmental Monitoring Programs," Ontario Power Generation.
- 20.8-7 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

## **20.9 Environmental Impact of Postulated Accidents Involving Radioactive Releases**

### **20.9.1 Design Basis Accidents**

The analysis of Design Basis Accidents (DBA) is described in Chapter 15, Subsections 15.7.7 and 15.7.8. Radioactive releases from the bounding DBAs analyzed are below the authorized limits. There are no DBAs that cause environmental releases from inside containment. Design Basis Accidents are outlined in Chapter 15, Subsection 15.5.4 for loss-of-coolant outside containment accident and Subsection 15.5.8 for fuel handling accident.

### **20.9.2 Design Extension Conditions**

The analyses of Design Extension Conditions (DEC) events without core damage are described in Chapter 15, Subsection 15.5.5. There are currently no DEC accidents described with core melting, as indicated in Chapter 15, Subsection 15.6.2. Radioactive releases from the bounding DEC events analyzed are below the authorized limits. There are no anticipated adverse impacts on the environment, as a result of the analyzed postulated initiating events.

### **20.9.3 Measures and Controls to Limit Adverse Impacts During Accidents**

Off-site monitoring of potential radioactive releases in accident conditions is discussed in Chapter 19, Section 19.3. The design of the BWRX-300 includes provisions to prevent and to mitigate the consequences of accidents and to ensure the likelihood is extremely low that an accident will have harmful consequences. The primary means of preventing and mitigating the consequences of accidents is through the application of Defence-in-Depth described in Chapter 3, Subsection 3.1.6.

### **20.9.4 Emergency Actions in Response to External Events**

The off-site emergency actions which are taken in response to an event at the station are discussed in Chapter 19. These off-site emergency actions are governed by and documented in N-PROG-RA-0001 Consolidated Nuclear Emergency Plan (Reference 20.9-1), which provides a written basis to document the concepts, roles, and resources required by OPG nuclear to implement and maintain its emergency response capability to protect the public, employees, and the environment in the event of a nuclear emergency. N-PROG-RA-0001 (Reference 20.9-1) provides a framework for interaction with external authorities and defines OPG's commitments under the Provincial Nuclear Emergency Response Plan.

### **20.9.5 References**

- 20.9-1 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.

## **20.10 Environmental Impact of Facility Decommissioning**

The information related to decommissioning available at this time is preliminary. Therefore, the assessment of potential environmental effects of decommissioning activities related to DNNP is presented at a conceptual level. The Preliminary Decommissioning Plan submitted to the CNSC separately describes the activities required to decommission the facilities associated with DNNP, and to restore the DNNP site for other OPG uses.

The decommissioning strategy for DNNP will be selected based on an assessment of relevant factors and will be documented in the Preliminary Decommissioning Plan. Decommissioning of the facility will occur in three phases: Transition from Operations to Decommissioning; Safe Storage and Monitoring (if required), and Dismantling, Disposal and Site Restoration (Reference 20.10-1). Chapter 21, Section 21.6 includes a discussion of the decommissioning activities expected for DNNP.

Prior to beginning decommissioning, a Detailed Decommissioning Plan and other documents will be prepared and submitted to the CNSC (Reference 20.10-1).

The primary concern during decommissioning is the potential release of radioactivity into the environment, including exposure to humans and non-human biota. During the Transition from Operations to Decommissioning phase, potential hazards are expected to be associated with residual radioactivity in systems and components; handling used fuel, filters, and resins; and decontamination activities. After approximately one year from shutdown, gamma radiation is expected to become the only residual external radiation hazard in the plant's vicinity other than the used fuel storage facility, due to decay of residual radiation fields. Radiation and industrial hazards during the Dismantling, Disposal and Site Restoration Phase are likely to be associated with dismantling, waste handling, transportation, and disposal (Reference 20.10-1).

The potential impacts from decommissioning to various environmental media are expected to be (Reference 20.10-1):

1. **Atmospheric Impacts:** Some dust and particulate matter may be released to the atmosphere during dismantling and demolition operations. Heavy equipment and vehicles used for transport of waste and other materials may release exhaust gases into the atmosphere. The nature and extent of these releases will depend on the type of equipment used and the intensity and duration of their operations. Construction equipment and blasting may result in increased noise levels during dismantling and demolition work. Good Industry Management Practices will be applied and can reasonably be expected to mitigate any adverse atmospheric effects of decommissioning beyond the Darlington Nuclear site.
2. **Surface Water Impacts:** A temporary increase in turbidity may result from filling and sealing the cooling water intake and discharge structures, as well as from runoff during dismantling and site restoration work. In addition, other dismantling activities may change the site topography and possibly increase site runoff. However, erosion control techniques and other practical measures are available which can effectively mitigate any adverse environmental effects. Good Industry Management Practices will be applied during decommissioning to ensure compliance with applicable water quality criteria, including treatment of surface water discharges, as required, before they are released to the environment.
3. **Aquatic Impacts:** Some increase in turbidity of the water along the shoreline may result from filling and sealing of the plant's cooling water intake and discharge structures as well as from runoff during the dismantling and site restoration work. Changes in runoff from

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

the site may potentially affect local aquatic life and thus may need to be mitigated. Erosion control and other practical measures will be applied to mitigate any adverse effects.

4. **Terrestrial Impacts:** The dust produced during demolition and site restoration work may temporarily affect some of the vegetation within the Darlington Nuclear site. These areas will be restored when the dismantling and cleanup is complete. Increased vehicular traffic during active phases of the decommissioning program, and associated traffic noise, activity, and vehicle/animal collisions, may also affect local wildlife. However, this disturbance would be temporary and wildlife populations would be expected to recover and return to the site at the end of decommissioning.
5. **Geological and Hydrogeological Impacts:** It is not expected that decommissioning activities will involve substantial alteration of the topography of the Darlington Nuclear site. Therefore, any potential effects of decommissioning on soil quality, groundwater quality and groundwater flow conditions will be bounded by the already-assessed effects of site preparation and construction.
6. **Land Use Impacts:** In general, it is assumed that any potential effects of DNNP decommissioning on adjacent off-site land uses will be bounded by the effects of site preparation and construction. Radioactive and hazardous wastes generated during decommissioning will be transported to appropriate off-site facilities for long-term storage or disposal. It is expected that non-radioactive waste from demolition of buildings and other structures will be recycled or reused, both on and off the site (e.g., concrete crushed and used for roadways, metal and wood components salvaged and/or remanufactured, etc.) or disposed of at appropriate off-site disposal facilities.

A number of activities will be undertaken during the decommissioning process to prevent environmental impacts from decommissioning (Reference 20.10-1).

1. **Transition from Operations to Decommissioning:** external non-fixed surface contamination will be removed from accessible areas of the station. Internal chemical decontamination will remove most of the radioactive material contained within contaminated systems. Most of the other hazardous, non-nuclear materials are removed. During this period, the normal station environmental surveillance and monitoring program are maintained.
2. **Safe Storage and Monitoring (if required):** Environmental surveillance and monitoring will be carried out during the safe storage period to ensure that potential releases of radioactive material to the environment are detected and controlled if they occur. Routine radiological monitoring of contaminated structures and systems will also be performed, as required, depending on the length of the Safe Storage and Monitoring activity. Procedures for responding to unanticipated changes in the radiological environment of the site and potential releases to the environment will be prepared and implemented if required.
3. **Dismantling, Disposal and Site Restoration:** By the end of this activity, the site will be free of industrial hazards. The site will be remediated and restored to a state suitable for other OPG uses, and it will meet the criteria established by the CNSC to release from regulatory control. Restoration work may include cleaning the site to remove any remaining inactive waste and debris, covering the site with gravel and topsoil and establishing a covering of vegetation to prevent soil erosion.

The DNNP decommissioning process will result in a staged reduction of material and radioactivity at the site. Accordingly, there will be a staged reduction in emissions and exposures to workers, the public and the environment. The radiological effects are expected to be well below applicable regulatory limits. Furthermore, radiological effects are expected to be substantially less than the



NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

effects associated with operation and maintenance of the DNNP station, which are already expected to be very low (Reference 20.10-1).

Decommissioning, when properly planned and carried out with effective control and mitigation, is likely to result in only minor adverse environmental effects. Specific effects which may eventually be caused by DNNP decommissioning will be assessed more fully as the beginning of decommissioning approaches (Reference 20.10-1).

**20.10.1 Reference**

20.10-1 NK054-REP-07730-00029, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.

## **20.11 Environmental Measurements and Monitoring Programs**

Environmental measurements and monitoring programs for activities related to the BWRX-300 SMR are discussed in the subsections below.

### **20.11.1 Darlington New Nuclear Plant Environmental Assessment Follow-up Program**

The Darlington New Nuclear Project Environmental Assessment Follow-Up Plan outlines the scope for environmental monitoring activities for the DNNP. The activities presented in the Environmental Monitoring and Environmental Assessment Follow-Up Plan address the DNNP Environmental Assessment follow-up monitoring requirements and associated baseline activities. Some of these activities will also be conducted to fulfill other requirements, such as monitoring required by CNSC licences; applicable federal, provincial, or municipal laws; and other monitoring as part of due diligence by OPG for the DNNP and overall Darlington Nuclear site. Monitoring activities outlined in the Environmental Monitoring and Environmental Assessment Follow-Up Plan include those that take place prior to site preparation and during the site preparation, construction, and operations phases of the DNNP (Reference 20.11-1).

The overall goal of the Environmental Monitoring and Environmental Assessment Follow-Up program is to confirm the validity of EIS conclusions, and the effectiveness of mitigation measures implemented. The data collected will provide the necessary information needed to determine the effectiveness of mitigation measures, as well as provide understanding of whether actual effects are equal to or less than those predicted (Reference 20.11-1).

OPG's Environmental Monitoring and Environmental Assessment Follow-Up program (Reference 20.11-1) for DNNP will be implemented under existing program governance within the Environmental Management System, OPG-PROG-0005 (Reference 20.11-2). The monitoring elements in the Environmental Monitoring and Environmental Assessment Follow-Up Plan (Reference 20.11-1) will be conducted in accordance with existing environmental monitoring programs at Darlington Nuclear, primarily Management of the Environmental Monitoring Programs, N-PROC-OP-0025 (Reference 20.11-3) and other programs such as Contaminated Lands and Groundwater Management, N-PROC-0044 (Reference 20.11-4).

Monitoring activities in the Environmental Monitoring and Environmental Assessment Follow-Up Plan are considered supplementary studies to the Environmental Monitoring Program and as such will be implemented through the existing Darlington Nuclear Environmental Monitoring Program. Once the objective of each supplementary study is achieved, a determination would be made, based on the results of the study, to either close off the study or integrate it into the routine Environmental Monitoring Program (Reference 20.11-1).

Activities under the Environmental Monitoring and Environmental Assessment Follow-Up Plan will be designed to meet one or more of the following objectives, as applicable by monitoring activity (Reference 20.11-1):

1. Identify adequate baseline characterization data for use in follow-up monitoring
2. Verify predictions of environmental effects identified in the environmental assessment
3. Determine the effectiveness of mitigation measures in order to modify or implement new measures where required
4. Support the implementation of adaptive management measures to address previously unanticipated adverse environmental effects

5. Provide information on environmental effects and mitigation that can be used to improve and/or support future environmental assessments including cumulative environmental effects assessments
6. Support environmental management systems used to manage the environmental effects of projects (i.e., DNNP activities)

The details for implementing the Environmental Monitoring and Environmental Assessment Follow-Up Plan activities are provided in associated monitoring plan/methodology reports.

#### **20.11.2 Environmental Monitoring Program**

The existing Darlington Nuclear Environmental Monitoring Program identifies the contaminants and physical stressors to be monitored for the operating OPG nuclear facilities and conducts monitoring in the environment surrounding the Darlington Nuclear site. The Environmental Monitoring Program design was developed using the guidance in CSA N288.4 (Reference 20.11-5) to address site-specific objectives covering the aspects of regulatory requirements, Environmental Risk Assessment (ERA) results, confirmation of effluent control, areas of regulatory interest, and stakeholder commitments. The Environmental Monitoring Program design uses a risk-based approach. Locations considered to be outside the influence of Darlington Nuclear site operations are also monitored to allow for a comparison with background values (Reference 20.11-6). The Darlington Nuclear Environmental Monitoring Program is governed by the Environmental Monitoring Program procedure, Management of the Environmental Monitoring Programs (N-PROC-OP-0025) (Reference 20.11-3).

The Environmental Monitoring Program is maintained in accordance with the DNGS operating licence. Environmental Monitoring Program reports are prepared and submitted to the CNSC in accordance with CNSC REGDOC-3.1.1 (Reference 20.11-7).

The Darlington Nuclear Environmental Monitoring Program is developed to satisfy the following primary objectives (Reference 20.11-6):

1. To assess the impact on human health and the environment of contaminants and physical stressors of concern resulting from operation of OPG nuclear facilities.
2. To demonstrate compliance with limits on the concentration and/or intensity of contaminants and physical stressors in the environment or assess their effect on the environment.
3. To demonstrate the effectiveness of containment and effluent control and provide public assurance of the effectiveness of containment and effluent control, independent of effluent monitoring.
4. To verify predictions made by ERAs, refine the models used in ERAs, or reduce uncertainty in the predictions made by ERAs.

Ontario Power Generation's existing environmental monitoring systems is comprised of on-site and site boundary monitoring system. Detailed information about environmental monitoring systems and sampling programs, including environmental sampling locations, sampling frequency, the number of samples taken, the media sampled, the sampling method, and the radionuclides monitored can be found in the 2021 Environmental Monitoring Program report (Reference 20.11-6), and are summarized in Chapter 2, Section 2.9. They consist of atmospheric, aquatic, and terrestrial sampling programs.

### **20.11.3 Effluent Monitoring Program**

The existing Darlington Nuclear Effluent Monitoring Program is governed by N-STD-OP-0031 Monitoring of Nuclear and Hazardous Substances in Effluents (Reference 20.11-8). This OPG standard outlines minimum requirements to establish an appropriate surveillance and monitoring program for nuclear and hazardous substances in airborne and waterborne effluents from operating OPG nuclear facilities, including DNGS, and will be extended to DNNP, under normal and abnormal operating conditions, in accordance with CSA N288.5 (Reference 20.11-9). The Darlington Nuclear Effluent Monitoring Program will be updated to encompass non-radiological releases from DNNP prior to fuel-out commissioning and will be further updated to encompass radiological releases from DNNP prior to DNNP operations.

The objectives of the Effluent Monitoring Program are to (Reference 20.11-8):

1. Demonstrate compliance with authorized release limits and any other regulatory requirements (e.g., action levels) concerning the release of nuclear and hazardous substances from the source
2. Demonstrate adherence to internal objectives and targets set on release amounts, for purposes of effluent control
3. Confirm the adequacy of controls on releases from the source
4. Provide an indication of unusual or unforeseen conditions that might require corrective action or additional monitoring
5. Provide data to assess the level of risk on human health and safety, and the potential biological effects in the environment of the nuclear and hazardous substances of concern released from facility
6. Confirm predictions in the EIS made through the Environmental Assessment process
7. Provide assurance to the public on the effectiveness of effluent control
8. Provide data which, when combined with the results of environmental monitoring and modeling, may be used to test or refine the models of the environment used in the ERA or dose assessments
9. Address any other objective identified by the site (e.g., demonstrating due diligence, meeting a stakeholder commitment, or other business reasons)

Documentation requirements for the effluent monitoring program are discussed in Section 20.12.

### **20.11.4 Groundwater Monitoring Program**

Ontario Power Generation has a groundwater monitoring program, which outlines directions and accountabilities for establishing and managing a groundwater monitoring program on OPG sites. This procedure discusses the program design, sampling and analysis requirements, and Quality Assurance/Quality Control requirements for the groundwater management program. It also covers the requirements for inspection and maintenance of wells, management of waste, reporting, program review, qualifications, and training.

Groundwater monitoring will be compliant with N288.7, Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills (Reference 20.11-10).

The groundwater monitoring wells are in key areas of the Darlington Nuclear site including the DNGS protected area (near the reactor buildings), controlled area (farther away from the reactor buildings but within the fence), and the site perimeter (Reference 20.11-11). Wells on DNNP lands are also sampled.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

The groundwater monitoring program monitors the on-site groundwater quality changes over time and examines the chemical, radiological, and physical characteristics of the groundwater beneath the site. The specific objectives of the groundwater monitoring program are (Reference 20.11-11):

1. Objective 1: Confirm predominant on-site groundwater flow characteristics
2. Objective 2: Monitor changes to on-site groundwater quality to ensure timely detection of inadvertent releases to groundwater
3. Objective 3: Ensure that there are no adverse off-site impacts from Darlington Nuclear groundwater

The design of the Darlington Nuclear groundwater monitoring program is risk-based in nature.

A groundwater monitoring report is written annually and is submitted to the CNSC and Ministry of Environment, Conservation and Parks, as required, based on previous site commitments with regulatory agencies (Reference 20.11-4).

#### **20.11.5 Monitoring Systems and Alarms**

As previously discussed in Chapter 11, Section 11.5, PREMS provides continuous monitoring to allow determination of the content of radioactive material in various gaseous, liquid process, and effluent streams. Inspection, testing and maintenance of PREMS is discussed in Chapter 11, Subsection 11.5.8.

Monitoring systems and components for the BWRX-300 plant are required to be redundant, diverse, independent, separate and of quality commensurate with safety category, seismic category, and quality category, as described in Chapter 3, Section 3.6. Seismic qualification of BWRX-300 equipment and associated systems are discussed in Chapter 3, Subsection 3.9.3. Environmental Qualification, discussed in Chapter 3, Subsection 3.9.4, is established to ensure that BWRX-300 Safety Class 1 Structures, Systems and Components (SSC) can perform their fundamental safety functions during and after exposure to a harsh environment resulting from a DBA during and after which they are required to operate. A preliminary safety classification list of SSC that are important control measures is in Chapter 3, Appendix 3A.

The alarm systems that are required for monitoring of unplanned radioactive releases are discussed in Chapter 11, Section 11.2 for LWM, Chapter 11, Section 11.3 for OGS, Chapter 9A, Section 9A.5 for the HVS and Chapter 11, Section 11.5 for PREMS.

#### **20.11.6 References**

- |         |   |
|---------|---|
| 20.11-1 | NK054-PLAN-07730-00014, "Environmental Monitoring and Environmental Assessment Follow-Up for the Darlington New Nuclear Project," Ontario Power Generation. |
| 20.11.2 | OPG-PROG-0005 "Environment Health and Safety Managed Systems," Ontario Power Generation.  |
| 20.11-3 | N-PROC-OP-0025, "Management of the Environmental Monitoring Programs," Ontario Power Generation.  |
| 20.11-4 | N-PROC-OP-0044, "Contaminated Lands and Groundwater Management", Ontario Power Generation.  |
| 20.11-5 | CSA N288.4, "Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills," CSA Group.                                       |

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

- 20.11-6 N-REP-03443-10027, "2021 Results of Environmental Monitoring Programs," Ontario Power Generation.
- 20.11-7 CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants."
- 20.11-8 N-STD-OP-0031, "Monitoring of Nuclear and Hazardous Substances in Effluents," Ontario Power Generation.
- 20.11-9 CSA N288.5, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills," CSA Group.
- 20.11-10 CSA N288.7, "Groundwater Protection Programs at Class I Nuclear Facilities and Uranium Mines and Mills," CSA Group.
- 20.11-11 NK38-REP-10140-10031 R001, "Darlington Nuclear Groundwater Monitoring Program Results," Ontario Power Generation

## **20.12 Records of Radioactive Releases and Availability of Information to the Authorities and the Public**

Records of routine radioactive releases from the site are made in accordance with the procedure for Management of the Environmental Monitoring Programs (N-PROC-OP-0025) (Reference 20.12-1) and Monitoring of Nuclear and Hazardous Substances in Effluents (N-STD-OP-0031) (Reference 20.12-2).

### **20.12.1 Record Storage and Retention**

As outlined in the Environmental Monitoring Program (EMP) procedure, reporting for OPG facilities shall be completed in accordance with the operating licence requirements. The procedure (Reference 20.12-2) refers to CNSC REGDOC-3.1.1 (Reference 20.12-3); Reporting Requirements for Nuclear Power Plants. The contents of the annual EMP report are to include, as a minimum, the reporting requirements defined in CNSC REGDOC-3.1.1, including:

1. A summary of the results of the environmental protection program and an analysis of the significance, with respect to the health and safety of persons and the protection of the environment, of the results of the Environmental Monitoring Program
2. The amount of nuclear substances, in International System of Units (SI), released to the environment and monitored as part of the effluent monitoring program, presented on an appropriate basis (weekly or monthly), along with a comparison to regulatory release limits
3. The amount of nuclear substances measured in the environment, in SI units, as part of the Environmental Monitoring Program
4. The results and calculations of the annual radiation doses to the representative persons and/or critical group or groups in comparison to the regulatory public dose limit with a description of the environmental transfer/exposure pathways associated with the operation of the nuclear power plant, including the dispersion and dosimetric models used
5. The amount of hazardous substances, in SI units, released to the environment and monitored as part of the effluent monitoring program, and measured in the environment as part of the Environmental Monitoring Program
6. For each parameter reported as part of the effluent/emission monitoring and EMP, a description of the characteristics of the monitoring results, including but not limited to the sample frequency (e.g., daily, monthly, semi-annually), sample type (e.g., grab, composite, activity counts over time), statistical quantity reported (e.g., weekly/monthly mean, annual average, annual total)
7. A description of any significant events, findings, or results, in respect to the conduct of the Environmental Monitoring Program
8. A summary of any proposed changes to the Environmental Monitoring Program

Ontario Power Generation will continue to meet the record keeping and reporting requirements of CNSC REGDOC-3.1.1 (Reference 20.12-3) through all phases of the DNNP.

The Environmental Monitoring Program procedure also outlines requirements for record storage and retention, including how records are to be filed, how document numbers are assigned, and the record retention period.

In accordance with the effluent monitoring standard (N-STD-OP-0031) (Reference 20.12-2), the minimum period of retention for sampling and monitoring records directly related to the effluent monitoring program shall span the pre-operational phase, operations phase, and decommissioning of the facility. For all other records not related to monitored data (e.g., training

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

records, sample collection, and analytical procedures), retention periods should be specified in the program documentation or in equivalent documentation, in line with Clause 10.4.3 of CSA N288.5-11 (Reference 20.12-4).

The procedure for Records and Document Management (OPG-PROC-0019) provides direction to ensure that records in the custody or control of OPG are consistently managed, protected, and accessible throughout their life cycle (Reference 20.12-5). These include higher-level requirements for management of records and documents, including environmental data records and documents as listed above. This procedure states that records shall comply with the requirements of legal and regulatory agencies, codes, standards, and corporate requirements that identify any records that need to be created or retained by OPG (Reference 20.12-5). Electronic records adhere to OPG-STD-0057, Electronic Document Management, which defines the requirements for managing electronic documents across OPG (Reference 20.12-6).

### **20.12.2 Data Availability to External Bodies**

OPG will continue to meet the record keeping and reporting requirements of CNSC REGDOC-3.1.1 (Reference 20.12-3) for the Darlington Nuclear site through all phases of the DNNP. Measures to make appropriate data available to authorities and the public will be consistent with operating OPG nuclear facilities.

As required in CNSC REGDOC-3.1.1 (Reference 20.12-3) the operating station shall submit routine scheduled reports on various topics that are required for longer-term compliance monitoring. The scheduled environmental reports specified in CNSC REGDOC-3.1.1 are (Reference 20.12-3):

1. Quarterly report on safety performance indicators (refer to Section 3.1 of CNSC REGDOC-3.1.1). In these reports, airborne and waterborne radiological release data are reported quarterly to the CNSC.
2. Annual report on environmental protection (refer to Section 3.5 of CNSC REGDOC-3.1.1). As prescribed in CNSC REGDOC-3.1.1, this report shall contain the results of the environmental protection program, the amount of nuclear and hazardous substances released to and measured in the environment, and the results of the annual radiation doses to the representative persons and/or critical groups.
3. Site environmental risk assessment (refer to Section 4.3 of CNSC REGDOC-3.1.1).

The document N-PROC-RA-0005 (Reference 20.12-7), Written Reporting to Regulatory Agencies, describes the processes for complying with regulatory requirements for written event reports to regulatory agencies and for scheduled reports to the CNSC. Regulatory Agencies include the following:

- Canadian Nuclear Safety Commission
- Ministry of Environment, Conservation and Parks
- Environment and Climate Change Canada
- Ministry of Labour
- Transport Canada (with respect to dangerous occurrences)
- Electrical Safety Authority

The Environmental Monitoring Program report for the Darlington Nuclear site shall be submitted to the CNSC by May 1st of the year following the reporting period. The annual Environmental Monitoring Program reports for Darlington Nuclear site should be posted to the OPG website in



NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

accordance with the Ministry of Energy's prescribed reporting dates for the year following the reporting period (Reference 20.12-1). Results of the DNNP EA follow-up monitoring program will be summarized in an annual report submitted to the CNSC. OPG also submits Environmental Compliance Approval annual reports and Emission Summary and Dispersion Modeling to the Ministry of Environment, Conservation and Parks prior to June 1st of each year (Reference 20.12-8). These reports are made available to the public upon request.

OPG's N-STD-AS-0013, Nuclear Public Information and Disclosure (Reference 20.12-9), documents OPG's process to meet regulatory and OPG disclosure requirements for effective and timely external communications to the public related to operations, health, safety, security, and the environment. This standard ensures that there are consistent standards and procedures for compliance with:

1. CNSC REGDOC-3.1.1 Reporting Requirements for Nuclear Power Plants (Reference 20.12-3)
2. ISO 14001 Standard requirements for communications with stakeholders and public (Reference 20.12-10)
3. Ontario Ministry of the Environment and Climate Change Environmental Compliance Approvals requirements regarding environmental complaints and concerns

Public information and disclosure, includes the provision to provide, in a timely and transparent manner, accurate information to stakeholders and the public in the vicinity of OPG's nuclear facilities regarding events, activities and operations. In terms of nuclear public information disclosure and transparency, OPG commits but is not limited to:

1. Report and post various emissions and spills on a quarterly and annual basis, depending on requirements, through established notification protocols with regulatory and municipal and provincial agencies
2. Post and communicate annual Environmental Monitoring Program reports
3. Post and communicate quarterly performance reports on facilities operations

Reports such as Environmental Monitoring Program reports, and Public Environmental Release Data Reports are posted online on a quarterly basis for public access, and information related to radioactive effluents and spills to the environment are made available to the public through OPG's website.

### **20.12.3 References**

- |         |  |
|---------|--|
| 20.12-1 | N-PROC-OP-0025, "Management of the Environmental Monitoring Programs," Ontario Power Generation.                 |
| 20.12-2 | N-STD-OP-0031, "Monitoring of Nuclear and Hazardous Substances in Effluents," Ontario Power Generation.          |
| 20.12-3 | CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants."                        |
| 20.12-4 | CSA N288.5, "Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills," CSA Group. |
| 20.12-5 | OPG-PROC-0019, "Records and Document Management," Ontario Power Generation.                                      |
| 20.12-6 | OPG-STD-0057, "Electronic Document Management," Ontario Power Generation.  |

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

- 20.12-7 N-PROC-RA-0005, "Written Reporting to Regulatory Agencies," Ontario Power Generation.
- 20.12-8 N-PROC-OP-0037, "Environmental Approvals," Ontario Power Generation.
- 20.12-9 N-STD-AS-0013, "Nuclear Public Information and Disclosure," Ontario Power Generation.
- 20.12-10 ISO 14001, Environmental Management Systems," International Organization for Standardization.

## **20.13 Appendix A: Annual Average Gaseous Effluent Releases for BWRX-300**

### **20.13.1 Methodology**

This analysis is performed with the GALE-BWR methodology in U.S. Nuclear Regulatory Commission (NRC) NUREG-0016 (Reference 20.13-1). The GALE-BWR method implements the assumptions outlined in NRC Regulatory Guide 1.112 (Reference 20.13-2). GALE-BWR calculates expected releases based on 1) standardized coolant activities derived from ANSI/ANS-18.1-2020 (Reference 20.13-3) Working Group recommendations, 2) release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, 3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environs, and 4) information received on the operation of nuclear power plants.

The calculations are based on data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment. The GALE-BWR method determines the release of radioactive materials (noble gases, particulates, carbon-14, tritium, argon-41, and iodine) in gaseous effluents from normal operation including anticipated operational occurrences from the following sources: main condenser OGS, turbine gland sealing system, mechanical vacuum pumps, and ventilation exhaust air from buildings in the BWRX-300 power block (the Reactor Building, Plant Services Area, Turbine Building, and Radwaste Building). For the purpose of this evaluation the BWRX-300 Plant Services Area is treated as an auxiliary building as defined in NUREG-0016 Revision 2. The BWRX-300 does not have a wetwell or suppression pool, so all of containment is a drywell as defined in NUREG-0016 Revision 2 (Reference 20.13-1).

The results determined in this analysis include:

- Annual airborne release for off-site dose evaluations
- Annual airborne release radionuclide totals
- Annual airborne release radionuclide concentrations

The operating data in NUREG-0016 is taken from large BWRs and is limiting for the much smaller BWRX-300 plant. As such, the data is subject to relaxations as newer data applicable to smaller BWRs is identified and validated for applicability to the BWRX-300.

### **20.13.2 Results**

The BWRX-300 annual average airborne release activities from various plant systems are shown in Table 20.13-1. The BWRX-300 total annual average airborne release activities and the airborne concentrations are shown in Table 20.13-2.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 20.13-1: BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr)**

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment
Kr-83m	-	-	-	-	-	-	2.2E-11	8.6E-03
Kr-85m	5.5E-03	1.6E-02	1.4E-01	-	-	-	2.1E-04	5.6E-04
Kr-85	-	-	-	-	-	-	6.0E+01	8.0E-03
Kr-87	-	1.1E-02	3.3E-01	-	-	-	6.0E-19	1.1E-03
Kr-88	5.5E-03	1.6E-02	5.0E-01	-	-	-	1.9E-07	1.4E-03
Kr-89	-	1.1E-02	3.2E+00	1.6E-01	-	-	-	8.7E-03
Xe-131m	-	-	-	-	-	-	1.0E+00	4.4E-03
Xe-133m	-	-	-	-	-	-	2.7E-08	8.3E-04
Xe-133	1.5E-01	4.5E-01	8.2E-01	1.2E+00	7.1E+00	-	6.9E-02	2.6E-02
Xe-135m	8.2E-02	2.5E-01	2.2E+00	2.9E+00	-	-	-	1.1E-03
Xe-135	1.8E-01	5.1E-01	1.8E+00	1.5E+00	2.7E+00	-	7.0E-50	2.2E-02
Xe-137	2.5E-01	7.4E-01	5.5E+00	4.5E-01	-	-	-	5.0E-04
Xe-138	1.1E-02	3.3E-02	5.5E+00	1.1E-02	-	-	-	5.2E-03
I-131	1.9E-04	9.9E-04	6.7E-03	4.2E-04	2.4E-03	6.0E-05	-	1.7E-04
I-132	2.1E-03	1.1E-02	7.2E-02	4.6E-03	-	-	-	3.0E-05
I-133	1.5E-03	7.4E-03	5.0E-02	3.2E-03	-	1.2E-04	-	1.9E-04
I-134	6.3E-03	3.2E-02	2.2E-01	1.4E-02	-	-	-	3.5E-05
I-135	2.9E-03	1.5E-02	9.9E-02	6.3E-03	-	-	-	1.2E-04

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment
H-3	1.3E+01	-	1.3E+01	-	-	-	-	-
C-14	-	-	-	-	-	-	1.5E+00	-
Na-24	-	-	-	-	-	-	-	4.1E-05
P-32	-	-	-	-	-	-	-	1.9E-05
Ar-41	-	-	-	-	-	-	8.8E-03	-
Cr-51	1.2E-06	5.5E-06	5.5E-06	4.3E-06	-	-	-	5.3E-04
Mn-54	2.4E-06	6.1E-06	3.7E-06	2.4E-05	-	-	-	3.1E-04
Mn-56	-	-	-	-	-	-	-	1.4E-05
Fe-55	-	-	-	-	-	-	-	6.6E-04
Fe-59	5.5E-07	1.8E-06	6.1E-07	1.8E-06	-	-	-	1.5E-04
Co-58	6.1E-07	1.2E-06	6.1E-06	1.2E-06	-	-	-	1.4E-04
Co-60	6.1E-06	2.4E-05	6.1E-06	4.3E-05	-	-	-	2.9E-04
Ni-63	-	-	-	-	-	-	-	6.6E-07
Cu-64	-	-	-	-	-	-	-	1.7E-04
Zn-65	6.1E-06	2.4E-05	3.7E-05	1.8E-06	-	-	-	1.4E-04
Rb-89	-	-	-	-	-	-	-	1.5E-06
Sr-89	1.8E-07	1.2E-07	3.7E-05	-	-	-	-	1.8E-06
Sr-90	1.8E-08	4.3E-08	1.2E-07	-	-	-	-	9.9E-08
Y-90	-	-	-	-	-	-	-	2.5E-08
Sr-91	-	-	-	-	-	-	-	4.8E-05
Sr-92	-	-	-	-	-	-	-	3.2E-05
Y-91	-	-	-	-	-	-	-	2.4E-05
Y-92	-	-	-	-	-	-	-	1.2E-05

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment
Y-93	-	-	-	-	-	-	-	3.7E-06
Zr-95	1.8E-06	4.3E-06	2.4E-07	4.9E-06	-	-	-	5.1E-05
Nb-95	6.1E-06	5.5E-05	3.7E-08	2.4E-08	-	-	-	4.7E-05
Mo-99	3.7E-05	3.7E-04	1.2E-05	1.8E-08	-	-	-	6.7E-05
Tc-99m	-	-	-	-	-	-	-	6.2E-06
Ru-103	1.2E-06	2.4E-05	3.1E-06	6.1E-09	-	-	-	1.2E-05
Rh-103m	-	-	-	-	-	-	-	4.9E-08
Ru-106	-	-	-	-	-	-	-	2.0E-06
Rh-106	-	-	-	-	-	-	-	6.5E-11
Ag-110m	2.4E-09	1.2E-08	-	-	-	-	-	6.5E-07
Sb-124	1.2E-07	1.8E-07	6.1E-07	4.3E-07	-	-	-	0.0E+00
Te-129m	-	-	-	-	-	-	-	2.3E-05
Te-131m	-	-	-	-	-	-	-	4.2E-06
Te-132	-	-	-	-	-	-	-	1.9E-06
Cs-134	4.3E-06	2.4E-05	1.2E-06	1.5E-05	-	-	-	1.8E-05
Cs-136	6.1E-07	2.4E-06	6.1E-07	-	-	-	-	9.7E-06
Cs-137	6.1E-06	3.1E-05	6.1E-06	2.4E-05	-	-	-	2.7E-05
Cs-138	-	-	-	-	-	-	-	3.3E-06
Ba-140	1.2E-05	1.2E-04	6.1E-05	2.4E-08	-	-	-	1.9E-04
La-140	-	-	-	-	-	-	-	1.9E-04
Ce-141	1.2E-06	4.3E-06	6.1E-05	4.3E-08	-	-	-	1.1E-05
Ce-144	-	-	-	-	-	-	-	2.0E-06
Pr-144	-	-	-	-	-	-	-	2.3E-09

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment
W-187	-	-	-	-	-	-	-	1.5E-05
Np-239	-	-	-	-	-	-	-	4.7E-05

(1) Cells marked with – represent no release.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 20.13-2: BWRX-300 Annual Airborne Releases and Concentrations Including AOOs**

<b>Nuclide</b>	<b>Airborne Release (Ci/yr)</b>	<b>Airborne Concentration (Ci/m3)</b>
Kr-83m	8.6E-03	3.6E-15
Kr-85m	1.6E-01	6.6E-14
Kr-85	6.0E+01	2.5E-11
Kr-87	3.5E-01	1.4E-13
Kr-88	5.2E-01	2.1E-13
Kr-89	3.4E+00	1.4E-12
Xe-131m	1.0E+00	4.3E-13
Xe-133m	8.3E-04	3.4E-16
Xe-133	9.8E+00	4.1E-12
Xe-135m	5.4E+00	2.2E-12
Xe-135	6.8E+00	2.8E-12
Xe-137	6.9E+00	2.8E-12
Xe-138	5.5E+00	2.3E-12
I -131	1.1E-02	4.5E-15
I -132	8.9E-02	3.7E-14
I -133	6.3E-02	2.6E-14
I -134	2.7E-01	1.1E-13
I -135	1.2E-01	5.1E-14
H-3	2.6E+01	1.1E-11
C-14	1.5E+00	6.1E-13
Na-24	4.1E-05	1.7E-17
P-32	1.9E-05	7.7E-18
Ar-41	8.8E-03	3.6E-15
Cr-51	5.5E-04	2.3E-16
Mn-54	3.5E-04	1.4E-16
Mn-56	1.4E-05	5.8E-18
Fe-55	6.6E-04	2.7E-16
Fe-59	1.6E-04	6.5E-17
Co-58	1.5E-04	6.2E-17
Co-60	3.7E-04	1.5E-16
Ni-63	6.6E-07	2.7E-19



NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Airborne Release (Ci/yr)	Airborne Concentration (Ci/m3)
Cu-64	1.7E-04	6.9E-17
Zn-65	2.1E-04	8.5E-17
Rb-89	1.5E-06	6.1E-19
Sr-89	3.9E-05	1.6E-17
Sr-90	2.8E-07	1.2E-19
Y-90	2.5E-08	1.0E-20
Sr-91	4.8E-05	2.0E-17
Sr-92	3.2E-05	1.3E-17
Y-91	2.4E-05	1.0E-17
Y-92	1.2E-05	5.1E-18
Y-93	3.7E-06	1.5E-18
Zr-95	6.2E-05	2.5E-17
Nb-95	1.1E-04	4.5E-17
Mo-99	4.8E-04	2.0E-16
Tc-99m	6.2E-06	2.6E-18
Ru-103	4.0E-05	1.7E-17
Rh-103m	4.9E-08	2.0E-20
Ru-106	2.0E-06	8.1E-19
Rh-106	6.5E-11	2.7E-23
Ag-110m	6.6E-07	2.7E-19
Sb-124	1.3E-06	5.5E-19
Te-129m	2.3E-05	9.3E-18
Te-131m	4.2E-06	1.7E-18
Te-132	1.9E-06	7.8E-19
Cs-134	6.2E-05	2.6E-17
Cs-136	1.3E-05	5.5E-18
Cs-137	9.5E-05	3.9E-17
Cs-138	3.3E-06	1.4E-18
Ba-140	3.8E-04	1.6E-16
La-140	1.9E-04	7.7E-17
Ce-141	7.8E-05	3.2E-17
Ce-144	2.0E-06	8.0E-19
Pr-144	2.3E-09	9.3E-22

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Airborne Release (Ci/yr)	Airborne Concentration (Ci/m3)
W-187	1.5E-05	6.1E-18
Np-239	4.7E-05	1.9E-17

### **20.13.3 References**

- 20.13-1 USNRC NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR)."
- 20.13-2 USNRC Regulatory Guide 1.112, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors."
- 20.13-3 ANSI/ANS-18.1, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society.

## **20.14 Appendix B: BWRX-300 Annual Average Liquid Effluent Activity Releases for BWRX-300**

### **20.14.1 Methodology**

This analysis is performed with the GALE-BWR 3.2 code described in U.S. Nuclear Regulatory Commission (USNRC) NUREG-0016 Revision 2 which implements the analysis methods and assumptions for liquid effluent releases in Regulatory Guide 1.112 (References 20.14-1 and 20.14-2).

The GALE code is a mathematical model for calculating the release of radioactive material in gaseous and liquid effluents from Boiling Water Reactors (BWR). The calculations are based on operating data from existing BWRs, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment.

GALE calculates expected releases based on:

1. Standardized coolant activities derived from ANSI/ANS-18.1-2020 standard (Reference 20.14-3) Working Group recommendations
2. Release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams
3. BWRX-300 LWM design features used to reduce the quantities of radioactive materials ultimately released to the environment
4. Operating data from existing BWRs

All liquid effluent releases from the BWRX-300 occur through the LWM.

### **20.14.2 Results**

The BWRX-300 annual average liquid effluent release activities from various waste streams are shown in Table 20.14-1.

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

**Table 20.14-1: Total Annual Average Liquid Effluent Release Including AOOs**

<b>Nuclide</b>	<b>Annual Release (Ci/year)</b>
I -131	1.8E-04
I -132	1.0E-05
I -133	2.0E-04
I -135	9.0E-05
Na-24	8.0E-05
P-32	3.0E-05
Cr-51	6.4E-04
Mn-54	4.0E-04
Mn-56	1.0E-05
Fe-55	8.0E-04
Fe-59	2.2E-04
Co-58	4.9E-04
Co-60	9.0E-04
Ni-63	9.0E-05
Cu-64	3.1E-04
Zn-65	9.8E-02
Zn-69m	1.7E-04
Br-83	6.0E-05
Sr-91	9.0E-05
Sr-92	3.0E-05
Y-91	3.0E-05
Y-92	9.0E-05
Zr-95	9.0E-05
Nb-95	1.3E-04
Mo-99	1.1E-04
Tc-99m	1.0E-04
Ru-103	2.0E-05
Ru-105	4.0E-05
Ru-106	4.5E-04
Ag-110m	6.0E-05
Te-129m	2.0E-05
Cs-134	6.0E-04

NEDO-33970 REVISION 1  
NON-PROPRIETARY INFORMATION

Nuclide	Annual Release (Ci/year)
Cs-136	6.0E-05
Cs-137	8.8E-04
Ba-139	1.0E-05
Ba-140	2.1E-04
Ce-141	3.0E-05
La-142	1.0E-05
Pr-143	2.0E-05
Ce-144	2.0E-04
W-187	3.0E-05
Np-239	8.0E-05

**20.14.3 References**

- 20.14-1 USNRC NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR)."
- 20.14-2 USNRC Regulatory Guide 1.112, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors."
- 20.14-3 ANSI/ANS-18.1, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society.



**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33971

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report**

**Chapter 21  
Decommissioning and End of Life Aspects**



**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
D&D	Dismantling and Demolition
DNNP	Darlington New Nuclear Project
GEH	GE Hitachi Nuclear Energy
HFEP	Human Factors Engineering Program Plan
LLW	Low-Level Waste
OPEX	Operating Experience
OPG	Ontario Power Generation
SSC	Structures, Systems and Components
SMR	Small Modular Reactor

## TABLE OF CONTENTS

<b>21.0 DECOMMISSIONING AND END OF LIFE ASPECTS.....</b>	<b>21-1</b>
21.1 Introduction .....	21-1
21.2 General Principles and Regulations.....	21-1
21.2.1 CSA Group.....	21-1
21.2.2 References.....	21-1
21.3 Incorporation of Decommissioning in the Design.....	21-2
21.3.1 Introduction .....	21-2
21.3.2 BWRX-300 SMR Decommissioning Design Elements.....	21-2
21.3.3 References.....	21-4
21.4 Long-Term Irradiated Fuel Storage.....	21-5
21.5 Decommissioning Strategy and Rationale .....	21-6
21.5.1 Objective of Decommissioning.....	21-6
21.5.2 Selection of Decommissioning Strategy .....	21-6
21.5.3 Description of Planned Decommissioning Activities .....	21-8
21.5.4 Transition to Decommissioning.....	21-11
21.5.5 Storage with Surveillance .....	21-12
21.5.6 Planning for Dismantling, Demolition and Site Restoration.....	21-12
21.5.7 Detailed Decommissioning Plan .....	21-14
21.5.8 Dismantling, Demolition and Site Restoration Period .....	21-15
21.5.9 Final End-State .....	21-17
21.5.10 Release from Regulatory Control.....	21-18
21.5.11 Reference .....	21-18
21.6 Human and Organizational Factors .....	21-19
21.6.1 References.....	21-20
21.7 Records.....	21-21
21.7.1 References.....	21-21
21.8 Waste Management.....	21-22
21.8.1 Waste Estimates .....	21-23
21.9 Financial Guarantee.....	21-24
21.9.1 References.....	21-25
21.10 Conclusion .....	21-26

**LIST OF TABLES**

Table 21.5-1: Conceptual DNNP Facility Decommissioning Schedule Compared to Decommissioning Schedule for Darlington Nuclear Generating Station and Darlington Waste Management Facility .....	21-10
---	-------

**LIST OF FIGURES**

Figure 21.5-1: Conceptual Planned Project Milestones Related to the Decommissioning of DNNP Facility.....	21-8
--	------

## **21.0 DECOMMISSIONING AND END OF LIFE ASPECTS**

### **21.1 Introduction**

Ontario Power Generation (OPG) is committed to effective and efficient life cycle management of nuclear facilities, which is in keeping with the evolution of international best practices in the nuclear industry. This chapter applies to decommissioning of the BWRX-300 facility at the end of its lifecycle.

The scope of decommissioning includes buildings used for operating the BWRX-300 at the Darlington New Nuclear Project (DNNP) site. These buildings include the Reactor Building, Turbine Building, Control Building, Radwaste Building, and the plant services area. Systems, components, piping, reactor vessel and reactor internals will be removed and appropriately packaged prior to disposal. Different decommissioning strategies are discussed, as well as the issues to be considered prior to determining the final decommissioning strategy. Human and organizational factors, records and financial guarantees are also discussed. Waste estimates will be provided in a separate submission to the Canadian Nuclear Safety Commission (CNSC).

### **21.2 General Principles and Regulations**

All decommissioning activities will be performed in compliance with relevant legislation, regulations, codes, and standards at the time of decommissioning.

The CNSC identifies the regulatory basis for decommissioning, as defined in the following key references:

- CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 21.2-1)
- CNSC REGDOC-2.11.2, "Decommissioning" (Reference 21.2-2)

#### **21.2.1 CSA Group**

In addition to the publications produced by the CNSC, decommissioning activities will also be performed in compliance with the CSA Group (CSA) standard below:

- CSA N294, "Decommissioning of Facilities Containing Nuclear Substances" (Reference 21.2-3).

#### **21.2.2 References**

- 21.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 21.2-2 CNSC Regulatory Document REGDOC-2.11.2, "Decommissioning."
- 21.2-3 CSA N294, "Decommissioning of Facilities Containing Nuclear Substances," CSA Group.

## **21.3 Incorporation of Decommissioning in the Design**

### **21.3.1 Introduction**

Operating Experience (OPEX) demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase. Assessment of future facility decommissioning and dismantling activities at the design phase includes consideration of OPEX gained from the decommissioning of existing facilities, as well as those facilities that are in long-term safe storage. The consideration of decommissioning at the design phase is expected to result in lower worker doses, reduced environmental impacts, and improved life cycle management of the facility (CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 21.3-1)).

The following are considerations of future plant decommissioning and dismantling activities, outlined in Section 7.24 of CNSC REGDOC-2.5.2 (Reference 21.3-1):

1. Materials are selected for the construction and fabrication of plant components and structures with the purpose of minimizing eventual quantities of radioactive waste and assisting decontamination.
2. Plant layout is designed to facilitate access for decommissioning or dismantling activities, including for plants with multiple units at a site, in periods when some units are operating, and some are under decommissioning.
3. Consideration is given to the future potential requirements for storage of radioactive waste generated as a result of new facilities being built or existing facilities being expanded.

### **21.3.2 BWRX-300 SMR Decommissioning Design Elements**

Decommissioning considerations made during the BWRX-300 Small Modular Reactor (SMR) design phase are discussed in this subsection.

#### **21.3.2.1 Site Plot Plan**

The BWRX-300 design has been optimized for constructability, which may be beneficial for dismantling the facility during decommissioning. Space allocations is considered to accommodate construction activities (i.e., access area around the plant, areas for laydown, etc.), thereby facilitating the ability to decommission and dismantle the plant once a Licence to Decommission the plant is granted.

#### **21.3.2.2 Modular Construction**

A modularization strategy will be used to design and construct the BWRX-300 at the DNNP site. The module and skid assemblies are intended to be built off-site, transported to the site, and erected on-site. This modularization strategy will provide guidance in selection of disassembly methods employed during decommissioning.

#### **21.3.2.3 Control of Materials During Design**

Specific guidance for plant systems materials to minimize corrosion products during plant operation is a design requirement, which provides restrictions regarding use of cobalt-based alloys and cobalt in stainless steel and nickel base alloys. These restrictions reduce personnel dose exposure during plant operation and decommissioning activities.

#### **21.3.2.4 Decommissioning Considerations for As Low As Reasonably Achievable**

Decommissioning considerations made to comply with the principle of As Low As Reasonably Achievable (ALARA), in the design of the BWRX-300, include the following aspects outlined in Chapter 12, Subsection 12.3.12:



NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

1. Shielding design considers protection during maintenance, inspection, decommissioning and operations.
2. To facilitate decommissioning, the Reactor Building, Turbine Building, and Radwaste Buildings are designed for large equipment removal, consisting of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings.
3. To facilitate decommissioning and ease of access, the radwaste process systems are skid-mounted and located in the Radwaste Building to allow truck access, and system skid loading and unloading.

**21.3.2.5 Design Features that Facilitate Decommissioning**

BWRX-300 design features that facilitate decommissioning by maintaining low occupational exposures are in Chapter 12, Subsection 12.1.5. Excerpts from Chapter 12, Subsection 12.1.5 are:

- Provisions for draining, flushing, and decontaminating equipment and piping
- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of piping systems
- Shielding that provides protection during maintenance, inspection, and operations, that may facilitate decommissioning
- Provision for adequate space for utilization of movable shielding
- Separation of more highly radioactive equipment from less radioactive equipment
- Provision for separate shielded compartments for adjacent items of radioactive equipment
- Provision for access hatches for the installation or removal of plant components
- Provision for the Reactor Water Cleanup System, Shutdown Cooling System, and the condensate demineralizer to minimize crud buildup

BWRX-300 design objectives that minimize radioactive contamination include:

1. Provide containment in areas where leaks and spills are most likely to occur.
2. Provide leak detection capability for prompt detection of leakage from Structures, Systems, and Components (SSC).
3. Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult (inaccessible) to conduct regular inspections (such as the Spent Fuel Pool and buried, embedded, or subterranean piping) to avoid release of contamination. BWRX-300 active liquid waste is kept within the radwaste building in tanks located in cubicles.
4. Facilitate decommissioning by minimizing embedded piping, sumps, or buried equipment.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

5. Design the plant to facilitate the removal or replacement of equipment or components during facility operation or decommissioning.
6. Minimize the generation of radioactive contamination and waste during operation and decommissioning by reducing the volume of components and structures that become contaminated during plant operation.

**21.3.3 References**

- 21.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

## **21.4 Long-Term Irradiated Fuel Storage**

The Nuclear Waste Management Organization is responsible for designing and implementing Canada's plan for the safe, long-term management of used nuclear fuel. The plan, known as Adaptive Phased Management, requires used fuel to be contained and isolated in a Deep Geological Repository. It also calls for a comprehensive process to select a site with informed and willing hosts for the project.

For used fuel storage, prior to moving the BWRX-300 fuel to a long-term Deep Geological Repository, the BWRX-300 design incorporates equipment and facilities necessary to load used nuclear fuel into containers for transfer to a licenced on-site independent spent fuel storage installation or licenced off-site waste management facility. Criticality safety requirements are considered throughout the life cycle of the fuel.

As part of the decommissioning plan, the last several fuel cycles of irradiated fuel will remain in the Spent Fuel Pool until they can be transported directly to the Nuclear Waste Management Organization Deep Geological Repository or stored in a licenced waste facility, until they can be accepted by the Nuclear Waste Management Organization. Spent fuel is an operational waste stream. The last cycle of irradiated fuel prior to plant shutdown will remain in the Spent Fuel Pool for a minimum of 2.5 years until it can be transported to a more permanent storage facility. Decommissioning activities can commence after used fuel is removed from the Spent Fuel Pool.

The 2.5-year period for storage of spent fuel in the Spent Fuel Pool is a theoretical minimum wet storage time. Factors that impact the spent fuel storage period include:

1. Burn-up time for fuel bundles
2. Spent Fuel Pool capacity
3. Type of flask used impacts the allowed minimum wet time as there should be consideration for both thermal rejection to the environment and radiation shielding

## **21.5 Decommissioning Strategy and Rationale**

The BWRX-300 SMR decommissioning strategy draws upon the lessons learned and experiences from decommissioning of similar reactors. With the removal of fuel from the reactor, activities associated with contaminated SSC are considered first when deciding on a decommissioning strategy.

In compliance with CNSC REGDOC-2.11.2, "Decommissioning" (Reference 21.5-1), BWRX-300 SMR decommissioning options considered include:

- Immediate (prompt) decommissioning
- Deferred decommissioning
- In-situ decommissioning

Note: In-situ decommissioning is not considered a reasonable decommissioning option for future facilities, except in exceptional circumstances, as per CNSC REGDOC-2.11.2 (Reference 21.5-1). In-situ decommissioning is not considered or evaluated for the DNNP facility.

### **21.5.1 Objective of Decommissioning**

The objective of the decommissioning program is to permanently retire the DNNP facility from service in a manner that will ensure the health, safety, and security of workers, the public and the environment are protected.

During decommissioning, radioactive and other hazardous materials, will be removed if present, and the DNNP facility site will be restored to meet the radiological release criteria approved by the CNSC. Upon completion of the decommissioning program, the DNNP facility will be in a condition that will support an application to the CNSC for release from regulatory control. OPG will retain ownership of the DNNP facility site, and it will then be available for other OPG uses.

The objective of decommissioning will be documented in the Preliminary Decommissioning Plan and will be reviewed and updated as necessary, with regular updates to the Preliminary Decommissioning Plan.

### **21.5.2 Selection of Decommissioning Strategy**

A decommissioning strategy will be selected early in the BWRX-300 SMR life cycle, to form the basis for planning for decommissioning and facilitate achieving the desired end-state of the decommissioning project. The following strategy options are considerations for selection individually or in combination:

1. Immediate (prompt) decommissioning:
  - a. To decontaminate, dismantle, or clean-up without any planned delays
2. Deferred decommissioning:
  - a. To place the facility, location, or site in a period of Storage with Surveillance (sometimes referred to as care and maintenance), followed by decontamination, dismantling, or clean-up
  - b. To conduct activities directed at placing certain buildings or facilities, locations, or sites in a safe and secure interim end-state, followed by a period of Storage with Surveillance, and ultimately, decontamination, dismantling and/or clean-up

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

3. In-situ decommissioning:

- a. To place the facility, location, site, or portions thereof, in a safe and secure condition in which some or all radioactive contaminants are disposed of in place, which may result in the creation of a waste disposal site

Note: As mentioned above, In-situ decommissioning is not considered or evaluated for DNNP.

Decommissioning strategies are evaluated in a systematic and traceable fashion so that the relative advantages and disadvantages of each strategy can be objectively compared. When determining the appropriate decommissioning strategy, the following are to be considered, as appropriate:

- Public and Indigenous engagement
- Potential impacts on Indigenous or treaty rights
- Operational experience and lessons learned
- Forms and characteristics of radioactive and non-radioactive contamination
- Integrity of containment and other SSC over time
- Availability of decontamination, disassembly, and clean-up technologies
- Potential for recycling or re-use of equipment and materials
- Availability of knowledgeable staff
- Potential environmental impacts
- Potential worker and public radiological doses
- End-state objectives and site redevelopment plans
- Potential revenues, costs, and available funding
- Availability of waste management facilities, locations, or sites
- Interdependencies with other facilities, locations or infrastructure located at the same site
- Assurance that the BWRX-300 site is maintained in a safe configuration
- Principles of radiation protection, justification, optimization, and application of dose limits

The DNNP facility decommissioning strategy will be documented in the Preliminary Decommissioning Plan and will initially be selected based on a high-level review of the above factors. In-depth studies will be performed as warranted over the life cycle of the plant, to refine and solidify the recommended decommissioning strategy as part of the regular review cycle to account for the following issues, which may have relevant consequences for decommissioning:

- Changes in site conditions, or incidents and events
- Changes to the proposed decommissioning objectives
- Changes to ownership or management structure
- Advances in decommissioning technology
- Significant modifications to the facility, location, or site
- Updated schedule, cost, and funding information

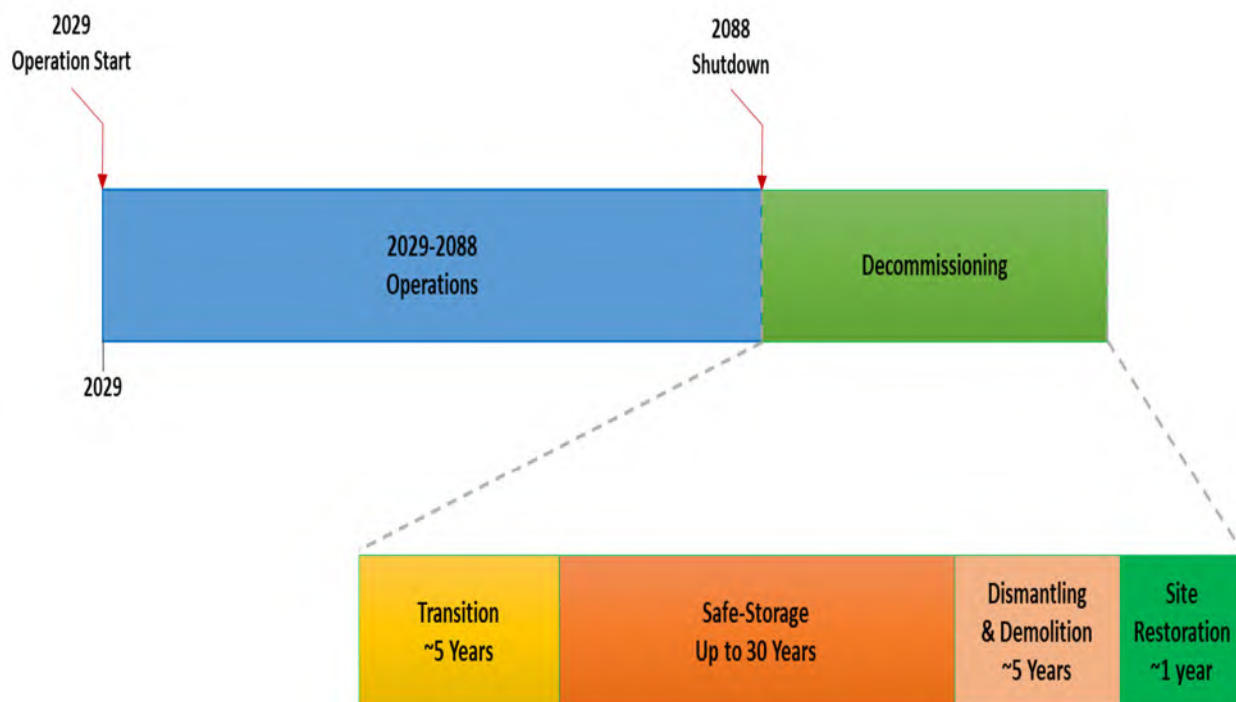
- Operational experience and lessons learned
- Revised regulatory requirements
- Availability of facilities, locations, or sites for the management of radioactive waste

Options for decommissioning are being considered and would be assessed as part of the Preliminary Decommissioning Plan.

### 21.5.3 Description of Planned Decommissioning Activities

This section outlines the major activities that will be performed during decommissioning of the BWRX-300 at the DNNP facility. The activities are described at a conceptual level with more detail being presented in the Preliminary Decommissioning Plan once it is prepared. All timing should be considered approximate and used for planning purposes only.

The anticipated major project milestones for decommissioning the DNNP facility are shown in Figure 21.5-1. More detailed schedules of decommissioning activities will be submitted to the CNSC as part of the Detailed Decommissioning Plan at the time of dismantling.



**Figure 21.5-1: Conceptual Planned Project Milestones Related to the Decommissioning of DNNP Facility**

OPG will prepare and submit a Detailed Decommissioning Plan to the CNSC for acceptance, two to five years prior to decommissioning, as part of OPG's application for a licence to perform decommissioning activities. The transition from operations to decommissioning is assumed to take five years once it has been shut down.

For planning purposes, a Storage with Surveillance period of up to 30 years will be assumed, followed by an assumed Dismantling and Demolition (D&D) period of five years. Site restoration and final surveys are assumed to take one year.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

Table 21.5-1 shows the conceptual DNNP facility decommissioning timelines compared to the Darlington Nuclear Generating Station and the Darlington Nuclear Waste Facility on the Darlington site, based on current planning assumptions for the three facilities.

The DNNP facility Preliminary Decommissioning Plan development ensures better alignment for future updates of the Darlington site decommissioning plans.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

**Table 21.5-1: Conceptual DNNP Facility Decommissioning Schedule Compared to Decommissioning Schedule for Darlington Nuclear Generating Station and Darlington Waste Management Facility**

<b>Phase</b>	<b>DNNP Facility (Conceptual Schedule Only)</b>	<b>Darlington Nuclear Generating Station</b>	<b>Darlington Waste Management Facility</b>
Shutdown	December 2088	Last Unit shut down December 2056	2082
Transition	Approximately five years	January 2057 – December 2057	Preparation for dismantling 2086 - 2089
Storage with Surveillance	Up to 30 years	January 2058 – December 2086 (Includes dry storage of used fuel at beginning of Storage with Surveillance and preparation for D&D at the end of Storage with Surveillance)	Not applicable
Dismantling and Demolition	Approximately five years	January 2087 – March 2089	2089
Site Restoration	Approximately one year	April 2089 – July 2092 (Includes final surveys)	2090



#### **21.5.4 Transition to Decommissioning**

OPG implements a program to incorporate industry OPEX and lessons learned from previous shutdowns and transitions to decommissioning as appropriate.

##### **21.5.4.1 Stabilization**

The transition, or Stabilization of the DNNP facility, will commence immediately following the end of commercial operations and be complete once the physical, operational, and administrative transition is confirmed.

Some of the key Stabilization activities are defueling and dewatering, radiation surveys, and removal of hazardous material. At the end of Stabilization, an Interim End-State Report will be produced and submitted to the CNSC to document the work that was executed.

The Stabilization goals and objectives include, but are not limited to:

- Defuel and removal of fluids, gases, hazardous materials, and other substances (e.g., resins) from the systems
- Continue to safely, and securely, store nuclear substances such as irradiated fuel until it can be removed to dry storage
- Maintain the facility in a safe and stable condition while creating no new hazards
- Reduce the footprint of the station in preparation for the next phase of decommissioning
- Protect workers, the public, and the environment from residual radioactive sources and hazardous materials remaining at the DNNP facility and maintain exposures ALARA

As Stabilization activities progress, systems that are no longer required to support the operation of the station will be placed into an inactive safe state. That is, they will be de-energized, drained of gas or fluids and isolated from operational systems. The operational, inspection, and maintenance requirements (as needed) for each system (or groups of related systems) will be identified and documented. Thereafter, the following activities will take place, as necessary:

1. Completing a Predictive Effects Assessment to proactively assess the potential environmental impacts resulting from the physical/operational changes to the station
  - a. Results of the assessment will identify environmental monitoring studies or mitigation measures required to manage the predicted effects.
2. Reviewing and revising operational programs to ensure requirements for the remaining phases of decommissioning are met
  - a. Examples include, but are not limited to environmental monitoring, radiation protection, emergency response, and fire protection. Plans and protocols, developed during the detailed planning phase, for monitoring the following would be submitted to the CNSC for acceptance:
    - i. Work hazards during decommissioning
    - ii. Personnel dosimetry
    - iii. Environmental emissions and effluents
    - iv. Materials, sites, and structures to be cleared from regulatory control

3. Developing a safety assessment framework to manage the nuclear and reactor safety aspects of Stabilization activities
4. Engaging stakeholders, including the public, and indigenous groups in transition planning activities

In support of transition to decommissioning, a safety assessment is also completed, including controls and approvals, to facilitate the shutdown and stabilization of the station. The objectives for the safety assessment include:

1. Demonstrate that applicable regulatory requirements are met throughout Stabilization
2. Demonstrate through systematic hazard analyses that the risks posed by hazards due to both Stabilization activities and for accident conditions are understood and managed
3. Identify necessary mitigating measures, limit controls and conditions to meet safety criteria throughout Stabilization
4. Quantify the hazard reduction to be achieved through Stabilization activities

#### **21.5.4.2 End-State of Stabilization**

By the end of the Stabilization period, all used fuel is planned to have been removed from the facility. The facility will remain intact with the structures and systems maintained in a safe condition. An Interim End-State Report indicating the current status of the facility will be prepared at the end of the Stabilization period for submission to the CNSC.

#### **21.5.5 Storage with Surveillance**

The Storage with Surveillance duration will be detailed in the Preliminary Decommissioning Plan and would delay the dismantling phase for a period of time. The Storage with Surveillance period allows for some decay of residual activation and fission products that remain in the station's systems prior to commencing D&D activities.

During Storage with Surveillance, SSC that would remain necessary to support continued operations (i.e., in active safe state) to meet operational demands will be modified or reconfigured, as necessary.

Programs will continue to support station operations and will be revised as necessary, similar to what is performed for Stabilization, as identified above. Programs and procedures will be adapted to meet regulatory requirements, while remaining commensurate with the complexity and risks of the Storage with Surveillance operations, and any revisions to these programs and procedures will require acceptance by the CNSC, where applicable.

During the Storage with Surveillance phase, OPG performs continuous monitoring and surveillance of the facility to ensure worker, public, and environmental safety is maintained.

An adequate level of security will be provided during Storage with Surveillance. Fire and radiation alarms are also to be monitored and maintained. Adequate plant staff will be available during this period to support the maintenance, inspection, and surveillance programs, as necessary.

#### **21.5.6 Planning for Dismantling, Demolition and Site Restoration**

Prior to the D&D phase, OPG will make detailed preparations for the dismantling, and the disposal of the waste. The Detailed Decommissioning Plan will be developed two to five years prior to dismantling operations and submitted to the CNSC for acceptance per regulatory requirements. A plan will be developed for the orderly progression D&D operations, including staff augmentation and any required plant system or service re-activation. Detailed work plans will be prepared to ensure that they remain appropriate with respect to improved knowledge of the condition of the

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

DNNP facility and the hazards that might be encountered during dismantling. Decommissioning Contractor(s) will be hired to manage and perform the D&D and Site Restoration. The activities performed by the Decommissioning Contractor(s) include, but are not limited to, updating procedures for characterization surveys, dismantling work, waste packaging, disposal, site restoration and final surveys. The D&D operations are designed to accomplish the required tasks while maintaining doses ALARA. Procedures will also address the continued protection of the health, safety, and security of workers, the public, and the environment.

During this phase of the work, OPG staff or Decommissioning Contractor(s) will take necessary actions, such as:

1. In consultation with the CNSC OPG will comply with applicable laws and regulations at the time of decommissioning, and will determine the requirements for performing an assessment of effects, to ensure that adequate provisions for the protection of the environment and the health and safety of persons are made prior to D&D
2. Develop a detailed schedule of activities with sequential planning of activities to minimize conflicts with simultaneous tasks
3. Reactivate, refurbish, or procure essential plant services necessary for dismantling
4. Review the results of the DNNP facility characterization and address any gaps or deficiencies in the information required to plan the decontamination, dismantling, demolition, and disposal
5. Prepare work packages for Decontamination, D&D, and disposal activities
6. Prepare detailed work procedures for the decontamination of SSC and procure decontamination equipment
7. Prepare detailed work procedures and sequences for the removal of systems and components
8. Perform a safety assessment to evaluate the processes for decontamination and dismantling the station, including waste handling, conditioning, and on-site processing
9. Evaluate the options for the disposal of the reactor vessel and its internals
10. Evaluate the options for the removal, handling, and disposal of other large radioactive components
11. Implement a graded approach for reviewing and revising station drawings, consistent with the need to maintain configuration control of the facility
12. Design, procure, and test the tooling and equipment (including remotely operated equipment) that will be used during the dismantling work
13. Procure needed D&D equipment
14. Procure or design and fabricate shielding and contamination control envelopes in support of removal and transportation activities
15. Develop procedures for occupational dose control, contamination control, industrial safety, environmental protection, fire protection, and emergency response
16. Develop or revise the emission monitoring program

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

17. Develop or revise the waste management program so that it covers the following processes, as applicable:
  - a. Characterization
  - b. Classification
  - c. Minimization
  - d. Segregation
  - e. Clearance
  - f. Handling
  - g. Volume reduction
  - h. Treatment
  - i. Packaging
  - j. Storage
  - k. Transportation
  - l. Final disposition
18. Develop a waste management plan, which typically includes but is not limited to:
  - a. Develop a plan for both the short term and, where possible, the long term, for management of all decommissioning waste
  - b. Develop the procedures for processing radioactive waste such as resins, filter media, and metallic and non-metallic waste generated during the dismantling work
  - c. Determine the transport and disposal container requirements for radioactive materials and hazardous wastes, including the requirements for shielding and stabilization of the waste
  - d. Procure and test the transportation and disposal containers for radioactive materials and hazardous waste
  - e. Prepare the detailed procedures for the packaging, removal, and disposal of radioactive materials, hazardous waste, and construction debris
  - f. Assess or investigate decontamination methods such as chemical cleaning, electro polishing, mechanical abrasion or melting. These decontamination methods may be used to decontaminate scrap metal if the reduction in the volume of the scrap is sufficient to justify further processing. Depending on the efficiencies achieved, metals will be considered as either radioactive wastes for controlled disposal, lightly contaminated (or activated) for consideration for re-use within the controlled nuclear environment, or metals that are decontaminated to levels below the clearance levels will be released for recycling in the open market
19. Prepare plans for site remediation
20. Obtain additional licences, permits, or approvals that are required and complete other regulatory processes that are applicable

#### **21.5.7 Detailed Decommissioning Plan**

A Detailed Decommissioning Plan is prepared and submitted for CNSC acceptance approximately two to five years prior to start of decommissioning. Towards the end of the

transition period, the Detailed Decommissioning Plan will be revised to describe OPG's detailed plan for managing the arrangements and activities conducted in support of the D&D and Site Restoration phases. The Detailed Decommissioning Plan will be prepared to meet the requirements of CNSC REGDOC-2.11.2 (Reference 21.5-1) and CSA N294, "Decommissioning of Facilities Containing Nuclear Substances" (Reference 21.5-2).

The Detailed Decommissioning Plan establishes criteria (i.e., clearance levels) that are used to determine if the material is suitable for uncontrolled release. The Detailed Decommissioning Plan also establishes the clearance levels and end-state criteria that are used to determine if the DNNP facility itself is suitable for release from further regulatory control.

The safety assessment for the DNNP facility decommissioning will be prepared in conjunction with the Detailed Decommissioning Plan, consider identified uncertainties, and address them as the decommissioning activities progress. Uncertainties regarding the radioactive inventory and the condition of the facility would be reduced as decommissioning progresses. The decommissioning safety assessment addresses potential radiological hazards to workers, the public, and the environment, from both routine decommissioning activities and credible accidents during decommissioning utilizing the defence-in-depth principle. The decommissioning safety assessment also identifies the mitigating methods to address the risks associated with these hazards and residual risks to the public once decommissioning is complete. The safety assessment will be reviewed, revised, or updated, as required, when additional information becomes available and as conditions change.

The environmental impact of facility decommissioning is discussed in Chapter 20, Section 20.10.

#### **21.5.8 Dismantling, Demolition and Site Restoration Period**

Dismantling work begins after collection of necessary permits and any other government approvals. Dismantling work has six conceptual steps:

1. Prepare the buildings and site
2. Decontaminate and dismantle nuclear systems
3. Dismantle non-nuclear systems
4. Demolish buildings
5. Manage and dispose of the waste
6. Restore the site

Work in the above steps may occur in parallel. Remedial action to support surveys for radioactive and other hazardous materials will be performed throughout the dismantling work, up to the final survey.

##### **21.5.8.1 Dismantling**

Dismantling activities are anticipated to involve the following:

1. Construct temporary facilities, modify existing storage facilities, and erect and place scaffolding in and around components to be dismantled to support the dismantling and decontamination activities. These may include a cutting station (as needed for large components), additional change rooms and contaminated laundry facilities for increased work force, establishment of laydown areas to facilitate equipment removal, upgrading roads to facilitate hauling and transportation, and modifications to the Reactor Building to facilitate access of large/heavy equipment.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

2. Modify containment to support segmentation activities and prepare rigging for segmentation and extraction of heavy components.
3. Conduct decontamination of components and piping systems as required to control (minimize) worker exposure. Remove, package and dispose of all piping and components that are no longer essential to support dismantling operations. It is anticipated that radioactive corrosion products on inner surfaces of piping and components will not have decayed to levels that will permit unrestricted use or allow conventional removal. These systems and components are surveyed as they are removed and disposed of in accordance with the radiological clearance levels that have been developed.
4. Remove systems and associated components as they become non-essential, such as related to decommissioning activities or worker health and safety. Remove systems (piping and components) utilizing contamination control practices, remote tooling, and packaging in approved transportation containers for disposal.
5. Remove reactor vessel and reactor internals. Package in Low-Level Waste (LLW) or Intermediate-Level Waste containers (as appropriate).
6. Remove activated and accessible contaminated concrete. Remove those portions of the associated enclosures necessary for access and component extraction.
7. Remove all remaining LLW and Intermediate-Level Waste along with any remaining hazardous materials. Material removed in the decontamination and dismantling of the nuclear unit will be routed to an on-site central processing area. Material that meets clearance criteria will be released for unrestricted disposition (e.g., as scrap, recycled or general disposal). Contaminated material will be characterized and packaged for controlled disposal at the long-term disposal facilities for respective LLW and Intermediate-Level Waste.
8. Remove remaining components, equipment including tools used in D&D, and plant services in support of the area release survey(s).
9. Conduct final radiation surveys to ensure that all radioactive materials in excess of permissible residual levels have been remediated.

Dismantling work performed on contaminated nuclear systems will be conducted in a manner that will minimize the spread of contamination and in accordance with OPG's Radiation Protection Program. Appropriate contamination control techniques, including the use of portal monitoring systems at controlled egress points, temporary enclosures, local ventilation, Personal Protective Equipment, and contamination monitoring, will be used when the work is performed.

#### **21.5.8.2 Demolition**

Once contaminated systems, structures and non-nuclear systems have been dismantled and final surveys have confirmed that the remaining structures are below radioactive and hazardous materials release limit, demolition activities may begin.

Efficient removal of the contaminated materials and verification that residual radionuclide concentrations are below regulatory limits will result in substantial damage to many of the structures. Remaining buildings and above ground structures will be removed using conventional demolition techniques.

Remaining structures (including buildings that were not contaminated and temporary structures) will be demolished by general demolition crews. Waste blocks will be sized so that they can be handled and moved by the available technologies. Foundation and exterior walls will be removed to the nominal one-meter removal depth below grade whenever possible. At-grade foundation

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

slabs exceeding one meter in thickness will be abandoned in place and covered with a one-meter-thick layer of backfill, as needed. Concrete rubble and clean fill produced by demolition activities may be used to backfill voids. Suitable materials will be used for filling, otherwise the rubble will be trucked off-site for disposal as construction debris.

Piping that exceeds the release criteria will be removed and disposed of appropriately. Clean metal piping will be considered scrap or will be recycled. Road and parking areas with asphalt or concrete surfacing are broken up and the rubble used for backfilling on-site if needed or disposed of appropriately.

Once demolition is complete, an Interim End-State Report will be prepared for submission to the CNSC. Waste management is discussed in Section 21.8.

#### **21.5.8.3 Restore the Site**

Depending on the nature of the future activities that will be carried out at the site of the DNNP facility, restoration work may include:

- Breaking up road and parking areas covered with asphalt or concrete surfacing and using the rubble for backfilling, if needed
- Cleaning the DNNP facility to remove any remaining inactive waste and debris
- Covering the filled excavations with gravel (for drainage) and topsoil
- Abandoning water drain holes at the bottom of all subgrade structures
- Restoring the lake front property (including the water inlets and outlets, as required) to inhibit erosion and potential detrimental impacts on fish, fowl, or other wildlife
- Grading the area to prevent ponding and inhibit the refloating of subsurface material
- Establishing a covering of vegetation to prevent soil erosion

Soil and structural surfaces within authorized limits may remain following D&D activities. The DNNP facility will be demonstrated to meet final clearance levels in support of release from further regulatory control.

#### **21.5.9 Final End-State**

The proposed end-state of the DNNP facility is that OPG will decommission and restore the site to an industrial end-state status suitable for other OPG uses, commonly known as “brownfield”. As per nuclear industry practice, a brownfield is defined as a former industrial land that has the potential to be developed for new industrial uses. By the end of the D&D and Site Restoration periods, the DNNP facility will be free of industrial and radiological hazards. Station SSC will have been dismantled.

Radioactive contamination in excess of established clearance levels and other hazardous materials will have been removed from the DNNP facility. The clearance level used for the clean-up of the facility should not require institutional controls after the release from regulatory control. Station systems will have been dismantled and buildings demolished. Subsurface structures will have been drained and de-energized. These subsurface structures will also have been surveyed for contamination, decontaminated, if required, and dismantled to a nominal depth of one meter below grade (consistent with international practices), backfilled with clean concrete rubble and soil, and graded over. The remaining facility will have been backfilled to prevent future subsidence and restored to a state suitable for other OPG uses. By the end of this phase, the end-state objectives defined in the Detailed Decommissioning Plan will be verified to have been achieved and the DNNP facility will meet the criteria for release from regulatory control.

#### **21.5.10 Release from Regulatory Control**

Upon completion of decommissioning, the DNNP facility will be in a condition to support its removal from regulatory control. A Final End-State Report (or site abandonment plan) on the decommissioning program will be prepared. The Final End-State Report will describe the decommissioning work that has been performed and the outcome of that work, the results of the final surveys that were performed, and the interpretation of those results (i.e., whether the results meet the end-state objectives defined in the Detailed Decommissioning Plan). Other information required by applicable regulations will also be included in the report. A Final End-State Report will be submitted to the CNSC as part of the request for a release from regulatory control. Institutional controls are not expected to be required.

#### **21.5.11 Reference**

- 21.5-1 CNSC Regulatory Document REGDOC-2.11.2, "Decommissioning."
- 21.5-2 CSA N294, "Decommissioning of Facilities Containing Nuclear Substances," CSA Group.



## **21.6 Human and Organizational Factors**

The period of transition from operations to decommissioning poses different Human and Organizational Factors challenges to OPG compared to the operational phase because of the complex process necessary to complete decommissioning.

OPG ensures Human and Organizational Factors issues are considered throughout the planning and execution of the project. Prior to decommissioning, OPG will establish a Human Factors Engineering Program Plan (HFEPP) for the decommissioning of the DNNP facility according to the guidance in CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 21.6-1); CNSC REGDOC-2.2.1, "Human Performance Management: Human Factors" (Reference 21.6-2); and lessons learned from decommissioning of Boiling Water Reactors, Pickering Nuclear Generating Station, and international OPEX. The details of the HFEPP will be included in the Detailed Decommissioning Plan. Special attention will be given to knowledge retention to minimize potential problems resulting from the loss of experienced personnel over time. Furthermore, a plan to manage staff reductions during the period following shutdown will be implemented. Risks associated with the decommissioning activities will be managed as per OPG's risk management process to ensure these risks are identified, analyzed, and documented and appropriate measures are put in place to mitigate these risks.

It is anticipated that the organization to oversee the decommissioning program will be assembled from available OPG station staff and outside OPG resources as needed. Decommissioning Contractor(s) will be hired to perform the D&D and Site Restoration work. OPG will provide the necessary oversight during this period. The Decommissioning Contractor(s) will be a company or consortium selected based on factors such as decommissioning experience, safety record, overall approach, and cost. OPG will remain the owner and licensee of the DNNP facility throughout the course of the decommissioning, but the Decommissioning Contractor(s) will be given charge and control of the facility during the D&D and Site Restoration periods. Other contractors may also be given charge and control of designated portions of the DNNP facility during the earlier phases of decommissioning. During these periods, the contractor will become the Constructor for the decommissioning work as defined by the Construction Safety Regulations made pursuant to the Occupational Health and Safety Act. The Decommissioning Contractor(s) and sub-contractors will be required to comply with OPG procedures related to Nuclear Energy Workers and other federal and provincial regulations.

OPG and the contractor(s) will ensure workers are qualified to perform the work assigned to them. Workers will be provided with training on the hazards associated with their work and the procedures that may be used to protect against those hazards. Training will include:

- Radiation protection
- Construction safety
- Workplace Hazardous Materials Information System
- Emergency Procedures
- Other appropriate topics

Staffing plans, which include clearly defined training requirements, are being developed by OPG to cover a 10-year rolling window and these plans will be updated annually. Staffing plans and associated training requirements will ensure training can be delivered within the specified timeframe to assure capability is maintained, and that sufficient resources are available for safe and reliable operation.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

Detailed descriptions of the required training for D&D will be included in the Detailed Decommissioning Plan that is expected to be available two to five years prior to entering the phase.

**21.6.1 References**

- 21.6-1 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 21.6-2 CNSC Regulatory Document REGDOC-2.2.1, "Human Performance Management: Human Factors."

## **21.7 Records**

Records filing and retention are governed by OPG's Information Management program, OPG-PROG-0001, "Information Management" (Reference 21.7-1), which identifies that records relevant to decommissioning are permanent records. Decommissioning-related documentation are managed and maintained in accordance with CNSC REGDOC-2.11.2, "Decommissioning" (Reference 21.7-2) and CSA N294, "Decommissioning of Facilities Containing Nuclear Substances" (Reference 21.7-3). These records will contain historical information that may be required in the future to update the Preliminary Decommissioning Plan, prepare the Detailed Decommissioning Plan, and ultimately facilitate successful decommissioning.

International decommissioning experience identifies retaining operational records improves future decommissioning. It is recognized that there is a potential for information about the DNNP facility to be lost as work transitions through the life cycle of the facility, and staff change, including the anticipated decrease in staffing numbers during the decommissioning period. It is therefore necessary that measures are taken to preserve records pertinent to decommissioning, capturing, and maintaining all potentially relevant information.

Records pertinent to decommissioning will be kept in the storage medium in standard use. All records will be assembled and maintained in accordance with the document and record management process and governance. The Preliminary Decommissioning Plan will document the operating and decommissioning records that will be retained, and the method of retention.

### **21.7.1 References**

- 21.7-1 OPG-PROG-0001, "Information Management," Ontario Power Generation.
- 21.7-2 CNSC Regulatory Document REGDOC-2.11.2, "Decommissioning."
- 21.7-3 CSA N294, "Decommissioning of Facilities Containing Nuclear Substances," CSA Group.

## **21.8 Waste Management**

The waste material inventory of the facility at the time of shutdown will depend on both the design and the operating history of the unit. The radioactive inventory will decrease over time due to the removal of activity by any decontamination work that is performed and the natural decay of the radioactive material. Estimates of the decommissioning waste inventory (i.e., the types and volumes of waste) will be prepared for the initial decommissioning plan supporting Licence to Operate and will be updated as needed based on survey and sampling information and operating history. Detailed characterization surveys will also be performed prior to the start of the D&D phase. The acquired characterization field data will permit an appropriate assessment of the radiological and conventional hazards that can affect workers, the public and the environment, and be used to develop the detailed decommissioning waste inventory for waste management planning.

Regarding disposal of radioactive decommissioning waste, OPG is exploring options after the cancellation of the Deep Geologic Repository for LLW and Intermediate-Level Waste at the Bruce Nuclear Site and remains committed to the permanent and safe disposal of its future decommissioning waste, including the DNNP facility waste.

OPG is also participating in Natural Resources Canada's work in public engagement on the existing Radioactive Waste Policy to ensure OPG is meeting international best practices. The Nuclear Waste Management Organization was asked to lead a dialogue to develop an integrated strategy for Canada's radioactive waste through close collaboration among waste owners and producers (including OPG), Indigenous people, and other interested Canadians. Progress on the Policy and Integrated Strategy will be taken into consideration in OPG's decommissioning waste disposal strategy.

For financial planning purposes, the LLW and Intermediate-Level Waste generated during decommissioning is assumed to be transferred to long-term disposal facilities for respective LLW and Intermediate-Level Waste.

Hazardous wastes generated during the D&D and Site Restoration periods of the decommissioning will likely be limited to hazardous materials originally used as building materials. Volumes of these wastes are likely to be small, since very few hazardous materials are expected to be used in the construction of the plant. Dry active waste such as combustibles (paper, cloth, wood, filter cartridges) could also be generated in the removal of plant systems.

The bulk of the non-hazardous waste materials generated during decommissioning will be produced during the D&D and Site Restoration periods of the decommissioning, although some are likely to be produced during the transition from operations period.

Appropriate disposal facilities for hazardous wastes will be identified prior to the beginning of the decommissioning project. Hazardous wastes will be packaged for transport and disposal according to the requirements of the applicable provincial regulations. All hazardous wastes, including nonradioactive hazardous wastes, will be transferred to an appropriate, licenced waste management facility for storage or disposal at approved disposal facilities. Waste manifests will be prepared and submitted as required by provincial regulations. Mixed waste (i.e., radioactive waste mixed with clean waste that is also hazardous) will be transferred to an appropriate long-term disposal facility.

If the volume or value of the contaminated scrap metal generated during decommissioning is sufficient to justify further processing, then chemical cleaning, electro polishing, mechanical abrasion, or melting might be used to decontaminate scrap metal. Metals that are decontaminated to levels below the clearance levels established in the Detailed Decommissioning Plan will be released for recycling or disposal.

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

Non-hazardous wastes that meet established clearance levels will be reused or recycled wherever possible or disposed of at approved disposal facilities. Clean concrete rubble may be used to fill the demolished DNNP facility or disposed according to applicable regulations. Other non-contaminated materials will be released for disposal according to the applicable regulations.

Existing site programs, instructions, and procedures will be evaluated to reflect the addition of the BWRX-300 plant for the handling of radioactive waste. Existing site programs, instructions and procedures will be evaluated and revised if necessary.

**21.8.1 Waste Estimates**

Volumes of radioactive decommissioning waste will be provided at a future licensing phase.

## **21.9 Financial Guarantee**

OPG established and maintains a financial arrangement that assures adequate funding of the decommissioning program (included in the Preliminary Decommissioning Plans) for the activities encompassed by the licence. The objective of OPG's Financial Guarantee is to ensure that sufficient funds are estimated, collected, and administered for the management of liabilities associated with operating and decommissioning of its nuclear facilities. The Financial Guarantee is prepared for OPG owned or leased facilities and makes specific financial provisions for the decommissioning of the facility. The facility's Preliminary Decommissioning Plan described in this Chapter forms the basis for establishing and maintaining an acceptable Financial Guarantee in support of the Licence to Operate.

In addition to the decommissioning program, OPG's Financial Guarantee also covers financial provisions for the long-term management (storage and eventual disposal) of operational and decommissioning wastes (e.g., Used Fuel, Low-Level and Intermediate-Level Wastes).

OPG's Financial Guarantee is prepared and maintained in accordance with the requirements set out in CSA N294, "Decommissioning of Facilities Containing Nuclear Substances" (Reference 21.9-1); CNSC REGDOC-2.11.2, "Decommissioning" (Reference 21.9-2) and CNSC REGDOC-3.3.1, "Financial Guarantees for Decommissioning of Nuclear Facilities and Termination of Licensed Activities" (Reference 21.9-3). OPG also provides an annual Financial Guarantee Report to the CNSC detailing the status of the guarantee, including the amounts accumulated in segregated funds and the value of the Provincial guarantee required. The Financial Guarantee Report compares the amount of the liabilities and the financial resources available to discharge obligations.

The Financial Guarantee provisions for the facility demonstrate that the current level of funding is adequate for decommissioning the station and returning the site to an end-state agreed upon with the Regulators. OPG will continue to provide annual Financial Guarantee reports to the CNSC detailing the status of the guarantee, including the amounts accumulated in segregated funds and the value of the Provincial Guarantee (if required).

The process described above is applicable for the Financial Guarantee for the facility prior to the start of operation. An appropriate financial instrument will be provided by OPG for the Financial Guarantee to meet the decommissioning liability for the construction phase of the DNNP project.

OPG reviews the Preliminary Decommissioning Plan as necessary, update the document to reflect new information and submits the Preliminary Decommissioning Plan to the CNSC for acceptance every five years, or as requested by the CNSC. The Preliminary Decommissioning Plan will be updated based on the following aspects, which can have consequences for decommissioning:

- Changes in site conditions, or incidents and events
- Changes to the proposed decommissioning objectives
- Changes to ownership or management structure
- Advances in decommissioning technology
- Significant modifications to the facility, location, or site
- Updated schedule, cost, and funding information
- Operational experience and lessons learned
- Revised regulatory requirements

NEDO-33971 REVISION 0  
NON-PROPRIETARY INFORMATION

- Availability of facilities, locations, or sites for the management of radioactive waste

OPG will progressively update the Preliminary Decommissioning Plan, where needed, to reflect the appropriate level of detail required for the licenced activities. Prior to the decommissioning phase, OPG will prepare a Detailed Decommissioning Plan.

**21.9.1 References**

- 21.9-1 CSA N294, "Decommissioning of Facilities Containing Nuclear Substances," CSA Group.
- 21.9-2 CNSC Regulatory Document REGDOC-2.11.2, "Decommissioning."
- 21.9-3 CNSC Regulatory Document REGDOC-3.3.1, "Financial Guarantees for Decommissioning of Nuclear Facilities and Termination of Licensed Activities."

### **21.10 Conclusion**

The BWRX-300 design incorporates design elements that allow for decommissioning. A conceptual facility decommissioning schedule has been prepared and will be revised prior to decommissioning of the BWRX-300 plant. As the design progresses, design elements that support decommissioning will be continually evaluated and incorporated. Options for decommissioning will be assessed as part of the Preliminary Decommissioning Plan. OPG maintains a Financial Guarantee for licenced activities in compliance with CNSC regulations.





**HITACHI**

**GE Hitachi Nuclear Energy**

NEDO-33972

Revision 0

September 30, 2022

*Non-Proprietary Information*

**Ontario Power Generation Inc.  
Darlington New Nuclear Project  
BWRX-300 Preliminary Safety Analysis Report:**

**Safeguards Annex  
Safeguards and Nuclear Material  
Accountancy**

**IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT**

**Please Read Carefully**

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

**REVISION SUMMARY**

<b>Revision #</b>	<b>Section Modified</b>	<b>Revision Summary</b>
0	All	Initial Release

### ACRONYM LIST

Acronym	Explanation
BWR	Boiling Water Reactor
CA	Complementary Access
CANDU	CANada Deuterium Uranium
CNSC	Canadian Nuclear Safety Commission
C/S	Containment and Surveillance
DIQ	Design Information Questionnaire
DNNP	Darlington New Nuclear Project
DNNP-1	Darlington New Nuclear Project, Unit 1
FPC	Fuel Pool Cooling and Cleanup System
GEH	GE-Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
IAEA	International Atomic Energy Agency
NSCA	Nuclear Safety and Control Act
NWMO	Nuclear Waste Management Organization
OPG	Ontario Power Generation Inc.
RB	Reactor Building
RPV	Reactor Pressure Vessel
SMR	Small Modular Reactor
TRS	Technical Report Series

## TABLE OF CONTENTS

<b>1.0</b>	<b>INTRODUCTION .....</b>	<b>SG-1</b>
1.1	Scope .....	SG-3
1.2	Relevant Legislations and Obligations .....	SG-3
1.2.1	Legislations and Regulations .....	SG-3
1.2.2	International Atomic Energy Agency Safeguards Agreements .....	SG-4
1.3	Limitations on Use .....	SG-5
<b>2.0</b>	<b>CLASSIFICATION AND CATEGORIZATION .....</b>	<b>SG-6</b>
2.1	Nuclear Material Classification.....	SG-6
2.2	Categorization of Licensees .....	SG-6
<b>3.0</b>	<b>SAFEGUARDS IN DESIGN.....</b>	<b>SG-7</b>
3.1	Design Information Requirements .....	SG-7
3.2	Design Information .....	SG-8
3.3	Design Information Examination and Verification.....	SG-9
<b>4.0</b>	<b>SAFEGUARDS AND OPERATION .....</b>	<b>SG-12</b>
4.1	Operational Information Requirements .....	SG-13
4.2	Operational Information .....	SG-13
4.3	Operational Information Examination and Verification .....	SG-14
<b>5.0</b>	<b>NUCLEAR FUEL LIFE CYCLE .....</b>	<b>SG-15</b>
5.1	Flow and Storage of Nuclear Fuel .....	SG-16
5.1.1	Fabrication and Transport.....	SG-16
5.1.2	Fresh Fuel Staging and Storage .....	SG-16
5.1.3	Core Fuel Shuffling/Refueling .....	SG-18
5.1.4	Irradiated Fuel Storage .....	SG-18
5.2	Nuclear Material Accountancy Software .....	SG-20
<b>6.0</b>	<b>ACCESS AND ASSISTANCE TO THE IAEA .....</b>	<b>SG-22</b>
6.1	Safeguards Equipment and Seals .....	SG-22
6.2	International Atomic Energy Agency Access.....	SG-22
<b>7.0</b>	<b>NUCLEAR MATERIAL ACCOUNTANCY .....</b>	<b>SG-24</b>
7.1	Measurement of Nuclear Material .....	SG-24
7.2	Provision of Information .....	SG-25
7.2.1	Nuclear Material Accountancy Reports .....	SG-25
7.2.2	Design Information.....	SG-26
7.2.3	Operational Information .....	SG-26

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

7.2.4	Information Required by the Additional Protocol .....	SG-27
7.3	Retention of Records .....	SG-27
<b>8.0</b>	<b>SAFETY, SAFEGUARDS AND SECURITY INTERFACES.....</b>	<b>SG-28</b>
<b>9.0</b>	<b>SAFEGUARDS, AND IMPORT AND EXPORT .....</b>	<b>SG-29</b>
<b>10.0</b>	<b>REFERENCES .....</b>	<b>SG-30</b>

**LIST OF TABLES**

Table 1.0-1: Selected Terms Used by the IAEA for Safeguards Purpose .....	SG-1
Figure SG-5.1.1-1: Top of Fuel Assembly .....	SG-21

**LIST OF FIGURES**

Figure SG-5.1.1-1: Top of Fuel Assembly .....	SG-21
---	-------



## 1.0 INTRODUCTION

Information is presented in this Safeguards Annex to demonstrate how Ontario Power Generation Inc. (OPG) supports compliance with the International Atomic Energy Agency (IAEA) safeguards agreement in connection with the Treaty (Reference SG.1) on the Non-Proliferation of Nuclear Weapons (Safeguards Agreement (Reference SG.2)) and the Additional Protocol (Reference SG.3) to the Safeguards Agreement (Reference SG.2). Under the Nuclear Safety and Control Act (NSCA) (Reference SG.4), the Canadian Nuclear Safety Commission (CNSC) has the mandate to achieve Canadian conformity with such international obligations.

Specifically, this Safeguards Annex describes information related to the BWRX-300 reactor facility at the Darlington New Nuclear Project (DNNP) to demonstrate compliance with the Safeguards Agreement (Reference SG.2) and the Additional Protocol (Reference SG.3), as well as compliance with the responsibilities included in the NSCA (Reference SG.4) and the General Nuclear Safety and Control Regulations (Reference SG.5).

This Safeguards Annex includes operational information based on OPG's Darlington Nuclear Generating Station practices. Although the high-level OPG safeguards program applies, the lower tier implementing documents are to be customized, as necessary, for the BWRX-300 Small Modular Reactor (SMR).

For clarification and consistency of terms used in this Safeguards Annex, Table 1.0-1 includes explanation of selected terms used by the IAEA for safeguards purposes.

**Table 1.0-1: Selected Terms Used by the IAEA for Safeguards Purpose**

(Source: IAEA Safeguards Glossary (Reference SG.10))

Term	Explanation
Containment	Structural features of a facility, containers or equipment which are used to establish the physical integrity of an area or items (including safeguards equipment or data) and to maintain the continuity of knowledge of the area or items by preventing undetected access to, or movement of, nuclear or other material, or interference with the items. Examples are the walls of a storage room or of a storage pool, transport flasks and storage containers. The continuing integrity of the containment itself is usually assured by seals or surveillance measures (especially for containment penetrations such as doors, vessel lids and water surfaces) and by periodic examination of the containment during inspection.
Surveillance	The collection of information through inspector and/or instrumental observation aimed at detecting movements of nuclear material or other items, and any interference with containment or tampering with IAEA equipment, samples, and data. Surveillance may also be used for observing various operations or obtaining relevant operational data. IAEA inspectors may carry out surveillance assignments continuously or periodically at strategic points.
Containment/ surveillance device (C/S device)	An item of equipment used to perform a C/S function and capable of providing its own C/S results

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

Term	Explanation
Containment/ surveillance measures (C/S measures)	<p>The application of containment and/or surveillance to complement nuclear material accountancy. The use of C/S measures is aimed at verifying information on movement of nuclear or other material, equipment and samples, or preservation of the integrity of safeguards relevant data. In many instances C/S measures cover the periods when the inspector is absent, thus ensuring the continuity of knowledge for the IAEA and contributing to cost effectiveness. Containment/surveillance measures are applied, for example:</p> <ul style="list-style-type: none"> <li>(a) During flow and inventory verification, to ensure that each item is verified without duplication and that the integrity of samples is preserved</li> <li>(b) To confirm that there has been no change to the inventory previously verified and thus reduce the need for remeasurement</li> <li>(c) To ensure that IAEA equipment, working papers and supplies have not been tampered with</li> <li>(d) If necessary, to isolate ('freeze') nuclear material that has not been verified until it can be measured</li> </ul>
System of containment/ surveillance measures (C/S system)	<p>A combination of containment and/or surveillance measures (see No. 8.6 [of Reference SG.10]). Each C/S system is designed to meet a purpose specified in the IAEA's safeguards approach. To increase reliability, a C/S system can include one or several C/S devices (see No. 8.3 [of Reference SG.10]). Containment/surveillance devices and containment may be used in such a way that each plausible diversion path (see No. 3.7 [of Reference SG.10]) is covered by at least one device (single C/S). For redundancy purposes, C/S devices may be backed up (duplicated) by a similar device. In a dual C/S system, each plausible diversion path is covered by two C/S devices that are functionally independent and are not subject to a common tampering or failure mode (dual C/S), e.g., two different types of seal, or seals plus surveillance. Dual C/S is normally applied where the verification of nuclear material is difficult to perform, in order to increase confidence in the C/S results and reduce the requirements for periodic reverification.</p>
Seal	<p>A tamper indicating device used to join movable segments of a containment in a manner such that access to its contents without opening the seal or breaking of the containment is difficult. A sealing system comprises the containment enclosing the material to be safeguarded, the means of applying the seal (e.g., a metal wire) and the seal itself. All three components must be examined in order to verify that the sealing system has fulfilled its function of ensuring continuity of knowledge of the identity and integrity of the material concerned. Seals in use by the IAEA include metal cap seals with tamper indicating features, as well as ultrasonic and electronic seals with fibre optic loops, and, for short time applications, tamper indicating paper tape seals. Sealing systems may be applied:</p> <ul style="list-style-type: none"> <li>(a) On safeguarded material or equipment to maintain the continuity of knowledge of the sealed contents between inventory verifications, and during shipment from one facility to another</li> <li>(b) On the operator's equipment (e.g., a crane) to monitor any use that would make possible the undeclared removal of nuclear material</li> <li>(c) On IAEA property (equipment, samples, standards, data, etc.) to prevent undetected tampering with it</li> </ul>

## **1.1 Scope**

The scope of this document covers safeguards information for OPG's BWRX-300 facility at the DNNP site, including:

1. Classification of nuclear material and categorization of licensees – Section 2.0
2. Safeguards design information – Section 3.0
3. Safeguards operational information – Section 4.0
4. Nuclear fuel life cycle – Section 5.0
5. Provision of access and assistance to the IAEA – Section 6.0
6. OPG's program on nuclear material accountancy – Section 7.0
7. Safeguards interface with safety, security – Section 8.0
8. Safeguards and Import and Export aspects – Section 9.0

## **1.2 Relevant Legislations and Obligations**

Subsection 1.2 describes the applicable legislations and regulations requirements and guidance related to safeguards, and the IAEA safeguards agreements.

### **1.2.1 Legislations and Regulations**

The following provisions of the NSCA (Reference SG.4) and the General Nuclear Safety and Control Regulations (Reference SG.5) made under it apply to this Safeguards Annex:

1. Subparagraph 9(a)(iii) of the NSCA establishes that one of the objectives of the CNSC is to "achieve conformity with measures of control and international obligations to which Canada has agreed."
2. Paragraph 24(4)(b) of the NSCA states that "[n]o licence shall be issued, renewed, amended, or replaced unless, in the opinion of the Commission, the applicant will, in carrying on that activity, make adequate provision for measures required to implement international obligations to which Canada has agreed."
3. Paragraph 26(a) of the NSCA states that "[s]ubject to the regulations, no person shall, except in accordance with a licence, (a) possess, transfer, import, export, use or abandon a nuclear substance, prescribed equipment, or prescribed information."
4. Paragraph 27(b) of the NSCA states that "[e]very licensee and every prescribed person shall (b) make the prescribed reports and file them in the prescribed manner."
5. Paragraph 12(1)(i) of the General Nuclear Safety and Control Regulations states that "[e]very licensee shall take all necessary measures to facilitate Canada's compliance with any applicable Safeguards agreement."
6. Section 30 of the General Nuclear Safety and Control Regulations identifies situations in which safeguards reports shall be provided by licensees to the Commission.

Subsection 4.13 of CNSC REGDOC-1.1.2 (Reference SG.15) requires the successful implementation of the obligations arising from the Non-Proliferation of Nuclear Weapons Treaty (Reference SG.1), Canada/IAEA Agreement (Reference SG.2), and the Additional Agreement (Reference SG.3), as well as the conformance with the requirements of the NSCA (Reference SG-4) and associated Regulations (Reference SG.5).

CNSC REGDOC-2.5.2 (Reference SG.14) requires, under Subsection 7.23, the BWRX-300 facility design and design process to allow for the permanent installation of IAEA safeguards equipment and the provision of services required for the ongoing operation of that equipment.

CNSC REGDOC-2.13.1 (Reference SG.6), on safeguards and nuclear material accountancy, sets out CNSC's requirements and guidance for the:

- Establishment and maintenance of a safeguards program
- Requirements for event and compliance monitoring reporting by licensees

Since the BWRX-300 is a SMR, CNSC REGDOC-1.1.5 (Reference SG.7) also allows for defining and using a graded approach commensurate with the risk when applying requirements and guidance contained in regulatory documents. However, it emphasizes that "[t]he use of a graded approach is not a relaxation of requirements, but rather the application of requirements in a manner commensurate with the risks and characteristics of a facility or activity."

This Safeguards Annex demonstrates how OPG satisfies the requirements of international obligations and regulatory requirements.

### **1.2.2 International Atomic Energy Agency Safeguards Agreements**

The Safeguards Agreement (Reference SG.2) came into force on February 21, 1972, while the Additional Protocol (Reference SG.3) became effective on September 8, 2000.

The Safeguards Agreement (Reference SG.2) describes the safeguards system to which Canada shall adhere, including what materials are subject to safeguards, and what information shall be reported to the IAEA. The objective of IAEA safeguards is "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection."

The Additional Protocol (Reference SG.3) covers further requirements for the provision of information and access to the IAEA, including the obligation to allow access to some locations on a 24-hour notice, and the obligation to provide information on and access to certain nuclear manufacturers and researchers, neither of which involve nuclear material. These tools allow the IAEA to have confidence that there is no undeclared nuclear material or nuclear activities in Canada, in compliance with the Safeguards Agreement (Reference SG.2).

IAEA guidelines for reactor facilities safeguards are offered in IAEA Technical Reports Series (TRS) No. 392 (Reference SG.8). It explains how information for new facilities is to be developed and proposes a schedule for submitting information to the IAEA. While these IAEA guidelines are applicable to the nuclear facility designs in the 1500 MWe size range, the report recognizes that smaller reactors passive safety features are being considered. On its website (Reference SG.9), the IAEA explains that "(t)here are challenges associated with performing nuclear verification at SMRs, including the use of new nuclear fuels, and irradiated fuel management considerations. However, by working with the IAEA at an early stage in reactor design, safeguards considerations can be embedded into the design of these reactors, so that nuclear verification activities are performed in the most effective and efficient way with minimal burden on the operator." Consequently, since the BWRX-300 is ~300 MWe SMR facility, it is possible that certain of the IAEA safeguards guidance would be implemented in a graded approach based on larger nuclear power plant designs addressed by the IAEA TRS No. 392 (Reference SG.8). This is in concert with CNSC REGDOC-1.1.5 (Reference SG.7).

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

The information presented in this Safeguards Annex demonstrates adherence to the obligations of the Safeguards Agreement (Reference SG.2) and the Additional Protocol (Reference SG.3), as well as in accordance with the guidance of IAEA TRS No. 392 (Reference SG.8).

### **1.3 Limitations on Use**

The Safeguards Annex is a high-level summary document developed to support the Licence to Construct activities for the DNNP-1 nuclear facility. The Safeguards Annex shall not be used for technical or construction purposes.

## **2.0 CLASSIFICATION AND CATEGORIZATION**

### **2.1 Nuclear Material Classification**

The BWRX-300 SMR facility utilizes Global Nuclear Fuels (GNF) GNF2 fuel. As listed in Chapter 1, Table 1.5-1, GNF2 fuel has an assembly-average enrichment of less than 5 per cent U-235. Such fuel is classified in CNSC REGDOC-2.13.1 (Reference SG.6) as Group 1A material, and thus is subject to full-scope safeguards of routine verification by the IAEA and detailed accountancy of nuclear material. Chapter 4, Subsection 4.2.2, and NEDO-33270 (Reference SG.17) provide detailed information regarding GNF2 fuel characteristics and design.

Before the initial nuclear fuel load and before each refueling cycle, the fresh fuel is shipped to the BWRX-300 facility and stored in the fuel pool, as described in Subsection 5.1.2. As listed in Chapter 1, Table 1.5-1, the irradiated fuel is stored in the fuel pool that is designed with a capacity for up to approximately eight (8) years of full operation. The irradiated fuel is then moved to a licensed on-site dry storage facility, as described in Subsection 5.1.4.2. The irradiated fuel will permanently be stored in a deep geological repository when it becomes available in Canada, as summarized in Subsection 5.1.4.5. Additional information on the fuel life cycle is found in Section 5.0.

Sensing, start-up or shielding equipment containing nuclear materials are managed, per CNSC REGDOC-2.13.1 (Reference SG.6), as Group 1A or Group 1B materials which also are covered under OPG's Safeguards and Nuclear Material Accountancy Program.

### **2.2 Categorization of Licensees**

Following the categorization of licensees described in Section 3.0 of CNSC REGDOC-2.13.1 (Reference SG.6), OPG is categorized as "facility for safeguards purpose" since the DNNP location will:

1. Include a reactor facility (that is, the BWRX-300 SMR) and a separate irradiated fuel dry storage installation
2. Be where Group 1A material in amounts greater than 1 effective kilogram are customarily used

### **3.0 SAFEGUARDS IN DESIGN**

The BWRX-300 facility design information includes:

1. Identification of the facility, stating its general character, purpose, nominal capacity and geographic location, and the name and address to be used for business purposes.
2. Description of:
  - a. The general arrangement of the facility, including site and building maps as needed, with reference to:
    - The form, location, and flow of nuclear material
    - The general layout of important items of equipment which are used to handle, produce, or process nuclear material
  - b. Features of the facility relating to nuclear material accountancy, as well as Containment and Surveillance (C/S)
  - c. Procedures used at the facility for nuclear material accountancy and control, with special reference to the structure of Material Balance Areas and Key Measurement Points, organizational responsibility, measurements of nuclear material, the random and systematic uncertainties of such measurements, and procedures for Physical Inventory Taking
3. Health and safety procedures

As the operator of the BWRX-300 facility, OPG ensures that the CNSC is in possession of up-to-date design information, specifically an IAEA Design Information Questionnaire (DIQ), the contents of which are described in Appendix I of IAEA TRS No. 392 (Reference SG.8).

#### **3.1 Design Information Requirements**

To enable planning its safeguards approach for a new reactor facility such as the BWRX-300 facility, the IAEA provides guidance in TRS No. 392 (Reference SG.8) for the establishment of a rolling program for the provision of design information. The information provided by such a program creates confidence in the peaceful purpose of the facility, and is intended to provide an adequate lead time to:

1. Ease the incorporation into the facility design, including the design of the nuclear material accountancy system and features which make it easier to implement safeguards at the facility
2. Allow time for safeguards research and development work that may be necessary
3. Enable the IAEA to carry out the budgetary planning necessary for effective and efficient implementation of safeguards
4. Permit the identification and scheduling of actions which need to be taken jointly by OPG and the IAEA, including:
  - a. Installation of safeguards equipment during construction of the facility
  - b. Verification of information on the design of the facility

The IAEA expects, on an iterative basis, information regarding safeguards relevant features of the facility design is made available early in the project definition, preliminary design, construction, and commissioning phases of BWRX-300.

During the project definition phase, the IAEA is provided with an initial DIQ, consisting of:

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

1. The identification of the facility and a short description of its general character, purpose, nominal capacity, and geographical location
2. The form, location, and movement of nuclear material
3. The general layout of important items of equipment which use, produce, or process nuclear material

During the subsequent preliminary design phase:

1. Discussion occurs with the IAEA on the safeguards approach to be used at the facility
2. Appropriate information on the design is provided to the IAEA as the design develops
3. Design incorporates the IAEA equipment or instrumentation required for safeguards purposes

The initial IAEA DIQ is based on the initial design and construction plan, and is provided to the IAEA, through the CNSC, at least 180 days prior to the start of construction. An update to the DIQ based on the "as-built" design and construction is then re-submitted, through the CNSC, to the IAEA approximately 180 days before the first receipt of nuclear material at the DNNP site. The CNSC coordinates with the IAEA and OPG for completion of the responses to the IAEA DIQ.

### **3.2 Design Information**

The initial DIQ, based on the summary of information requested in Appendix I of TRS No. 392 (Reference SG.8), is developed for the BWRX-300 for submission to the CNSC no later than 270 days before the start of construction. Note that CNSC recommends that the applicant submit the DIQ either before or concurrent with the Licence to Construct application (refer to Subsection 4.13.1 of CNSC REGDOC-1.1.2 (Reference SG.15)). The DIQ is a separate, non-public submittal.

The BWRX-300 DIQ is based on safeguards relevant preliminary design information of the BWRX-300 facility, as described in:

- Chapter 1, Section 1.5 for general BWRX-300 Power Block description
- Chapter 2, Subsection 2.1.2 for site and facility layouts
- Chapter 3, Subsection 3.5.5 for Reactor Building (RB)
- Chapter 4, Sections 4.2 and 4.3 for fuel system and nuclear design
- Chapter 5, Sections 5.4 for Reactor Pressure Vessel (RPV) and internal components
- Chapter 9A, Section 9A.1 for fuel storage and handling systems
- Chapter 9B, Section 9B.2 for integrated RB design
- Chapter 15, Subsections 15.5.7 for fuel pool deterministic safety analysis and the results thereof

Design information requested by the IAEA (Reference SG.8) to be reported in the BWRX-300 DIQ includes:

1. General information about the facility, such as location, owner, operator, purpose, main features, schedules for construction/commissioning/operation, operating modes, site layout, and reactor facility layout



NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

2. Reactor data, such as general flow diagrams, thermal and electricity outputs, number of reactor units, type of reactor and of fueling, core enrichment, reactor coolant and moderator
3. Description of the nuclear material, such as the fresh fuel and its enrichment, nominal weight of fuel in element/assemblies, physical and chemical form of the fresh fuel
4. With relevant drawings attached under reference numbers, description of reactor assemblies, and fresh fuel elements
5. Description of elements exchange in assemblies, basic accounting unit, and means of nuclear material identification, as well as other nuclear material in the facility
6. Information on nuclear material flow, such as:
  - a. Diagrams attached under reference numbers identifying, for example, Material Balance Areas, Key Measurement Points
  - b. Inventory and locations of fresh fuel, in-core fuel, irradiated fuel, refueling, burnup, storage of irradiated fuel
  - c. Reactor load factor, core loading, refueling, and burnup
  - d. Whether the irradiated fuel is processed or stored
7. Handling of fresh and irradiated fuel information, including, fresh fuel transport and storage, locations in reactor vessel, transfer to fuel pool, nuclear material testing areas, relevant instrumentation to measure safeguards key parameters
8. Reactor and fuel pool coolant data, such as mass flow, temperature, and pressure at major or key locations
9. Information on nuclear material accountancy and control
10. Measures to protect and secure nuclear material, confine, and contain safeguards equipment, and protect health and safety of inspectors during compliance
11. Other measures that are deemed necessary or specific for the BWRX-300 facility or the DNNP site

Information included in the DIQ is used by the IAEA to prepare the safeguards approach and to serve as the basis for preparing a draft for inspection goals and procedures, including planning for site visits, inspections, and audits.

### **3.3 Design Information Examination and Verification**

The IAEA verifies design information through the physical examination of the BWRX-300 facility during construction, commissioning, operation, and subsequent phases.

A design information examination is a review to determine whether further information is necessary for the IAEA to establish adequate safeguards and to identify features of the facility and the nuclear material handling and storage procedures which may have an impact on the safeguards approach and methods.

A Design Information Verification is an on-site verification by the IAEA to ensure that the information is sufficiently complete and correct and satisfies IAEA needs as specified in the Safeguards Agreement (Reference SG.2). This verification is performed during a visit, which is different than other IAEA inspections that are arranged for specific purposes.

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

Before an on-site visit, the IAEA performs specific actions such as:

1. Examining the design information provided in the DIQ to establish its completeness and its self-consistency
2. Determining whether any additional information or clarification is required that could have an effect on safeguards; and requesting formally such additional information from the CNSC, as needed
3. Preparing a preliminary safeguards approach for the BWRX-300 facility, based on the design information included in the initial DIQ and other relevant information available which considers specific features of the facility and other relevant features and elements indicated in the Safeguards Agreement including:
  - a. Determination of possible diversion strategies, identification of possible diversion paths, and specification of the detection goals (detection times, significant quantities, and detection probabilities) to be achieved for the BWRX-300 facility
  - b. Analysis of the identified possible diversion paths
4. Preparing the draft Facility Attachment (employing the model Facility Attachment described below), the design information, the goals and procedures and any other relevant facility information
5. Planning the on-site verification activities

A model for the Facility Attachment, to be prepared by the IAEA, is proposed in IAEA TRS No. 392 (Reference SG.8) to contain, among other things:

1. A short description of the facility
2. A provision to submit to the IAEA changes in the information on the facility
3. The accountancy measures to be used at the facility
4. Provisions for the C/S measures
5. Specific provisions and criteria for the termination of and exemption from safeguards of nuclear material
6. Description of the records and reports system
7. Description of the mode and the scope of IAEA routine inspections
8. Provisions for administrative and financial procedures concerning the application of safeguards at the facility

During the IAEA on-site visit to perform verification of the design information prior to and/or during construction, the IAEA will, for example:

1. Identify in sufficient detail those features affecting the application of safeguards to ensure that the safeguards objectives, as laid down in the Safeguards Agreement (Reference SG.2) are met and the IAEA safeguards goals are fulfilled
2. Determine the Material Balance Areas used for IAEA accounting purposes and select Key Measurement Points used to determine the flow and inventory of the nuclear material. When determining a Material Balance Area, the following criteria are considered:
  - a. The size of any Material Balance Area is based on the material balance accuracy obtainable

NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

- b. The use of C/S measures to ensure the completeness of flow measurements, and to simplify the application of safeguards and focus measurement efforts at the Key Measurement Points
- c. The number of Material Balance Areas at the BWRX-300 facility or at other defined locations which may be combined into one Material Balance Area if the IAEA determines that this is feasible
- d. Establishment of:
  - i. A special Material Balance Area, at the request of the CNSC, around a process step involving commercially sensitive information
  - ii. Nominal timing and procedures for taking physical inventory for IAEA accounting purposes
  - iii. Requirements for records and reports and the evaluation procedures
  - iv. Requirements and procedures for flow and inventory verification, specifying quantities and identifying the locations of nuclear material
- e. Selection of appropriate combinations of C/S methods and techniques and the strategic points at which they are to be applied

Furthermore, during the IAEA on-site visit and to the extent practicable, the planned IAEA safeguards approach and inspection goals as well as procedures are checked against the design information and any additional information received on BWRX-300 practices or "State System Accounting and Control" activities (which means, in this Safeguards Annex CNSC activities). Features of the BWRX-300 that are of special interest include:

1. C/S features and provisions
2. Operational practices, including Material Balance Areas and Key Measurement Points established by OPG and as used by the CNSC
3. The flow of nuclear material, its type, and its physical and chemical form (refer to Section 5.0)
4. Possible identification methods of items and batches
5. Measurement methods used and their accuracy
6. Sampling procedures and methods for ensuring the integrity of samples
7. Methods of storage and containment of nuclear material from the viewpoint of measurement techniques used by the IAEA
8. Recording and reporting procedures
9. Possibilities of using C/S equipment

As the operator of the BWRX-300, OPG responds promptly to IAEA requests for additional information on the IAEA examination of the design information contained in the DIQ, as well as on operational information relevant to safeguards. OPG assists and provides access to IAEA personnel to facilitate the IAEA planned or supplemental on-site visits. Additional information on these aspects is found in Sections 4, 6, and 7.

## **4.0 SAFEGUARDS AND OPERATION**

Section 4 of CNSC REGDOC-2.13.1 (Reference SG.6) specifies that the licensee (that is OPG) maintains a documented Safeguards and Nuclear Material Accountancy Program which fulfills applicable safeguards requirements. For the BWRX-300 facility and locations outside BWRX-300 facility as well as other facilities, OPG has a safeguards program (Reference SG.16) which meets the applicable safeguards requirements as follows:

- Safeguards Equipment and Seals (Subsection 6.1)
- IAEA Access (Subsection 6.2)
- Nuclear Material Accountancy (Section 7.0)
- Provision of Information (Subsection 7.2)
- Retention of Records (Subsection 7.3)

The health of the program is reviewed and assessed quarterly and annually by OPG using a set of Key Performance Indicators. Areas of the program that require performance improvement are routinely listed and tracked to completion using action plans.

The Safeguards and Nuclear Material Accountancy Program establishes OPG's implementing and reporting processes and procedures that are used to ensure Canada's fulfillment of its obligations for Nuclear Material Accountancy. The program outlines roles and responsibilities for individuals and organizations mandated to meet the requirements and guidance of CNSC REGDOC-2.13.1 (Reference SG.6).

The current OPG Safeguards, and Nuclear Material Accountancy Program applies to the following five OPG "Facilities", as per the classification and categorization of Section 2.0:

- Darlington Nuclear Generating Station
- Pickering Nuclear Generating Station
- Nuclear Sustainability Services Darlington (used fuel storage)
- Nuclear Sustainability Services Pickering (used fuel storage)
- Nuclear Sustainability Services Western (used fuel storage)

Each of these facilities is designated as a Material Balance Area for which a dedicated Safeguards Officer is responsible for:

- Accountancy of nuclear materials
- Facilitating IAEA access
- Maintaining provisions of information
- Retention of records

OPG employees who access areas or work around areas with IAEA C/S equipment are trained to identify such equipment and to not manipulate the equipment without appropriate notification.

OPG Safeguards Officers or Delegates, in collaboration with OPG operations and maintenance organizations, facilitate and provide access to the IAEA for Unannounced Inspections, Short Notice Random Inspections, planned inspections or maintenance, and for Complementary Access (CA) requests as required by CNSC REGDOC-2.13.1 (Reference SG.6).

#### **4.1 Operational Information Requirements**

The OPG Safeguards and Nuclear Material Accountancy Program (Reference SG.16) meets the requirements in Section 4 of CNSC REGDOC-2.13.1 (Reference SG.6). This is achieved through:

1. Strict inventory tracking of nuclear materials by using a nuclear material accountancy software
2. Training of staff involved with the maintenance, inspection, verification, or movement of nuclear materials on the requirements of reporting and tracking information using either software or physical means
3. Site specific operating procedures that provide instructions on meeting the established safeguards agreement
4. Strict access control to areas that contain nuclear materials
5. Clearly identifying facility personnel with safeguards roles and responsibilities

The nuclear material accountancy software provides the Safeguards Officers tools to generate reports required in Section 8 of CNSC REGDOC-2.13.1 (Reference SG.6).

Furthermore, OPG submits documents to facilitate Canada's fulfillment of its Safeguards and Nuclear Material Accountancy obligations. Safeguards related information is maintained on the Nuclear Material Accountancy Software, which provides near real-time inventory of:

- Fresh and irradiated fuels
- Fuel that is in the reactor core
- Fuel that is located anywhere else within the facility to support operations
- Other non-fuel nuclear materials required to support operations

OPG's Safeguards and Nuclear Material Accountancy Program (Reference SG.16) provides direction for Site Safeguards Officers to perform their day-to-day duties. To support submission of required documents listed in Appendix D: Table D1 – Reporting Frequency for Nuclear Material Accountancy Reports of CNSC REGDOC-2.13.1 (Reference SG.6), OPG maintains an action tracking system to ensure that documents are completed and appropriately submitted.

#### **4.2 Operational Information**

OPG Safeguards Officers submit to the CNSC by September 30<sup>th</sup> of each year an operational program for the BWRX-300 facility (refer to Subsection 8.3 of CNSC REGDOC-2.13.1 (Reference SG.6)). This operational program is updated to ensure it remains current. This operational program outlines:

1. Preferred date of the Physical Inventory Taking for the upcoming calendar year
2. Anticipated shutdown periods during the upcoming calendar year
3. Information on expected transfer of nuclear materials in the next calendar year
4. Updates on current or upcoming projects of relevance to safeguards

These annual operational program submissions are traced through action tracking software that requires, prior to closing out an action, closure notes confirming that the action has been completed. Additional information about the nuclear material accountancy software and inventory tracking is found in Subsection 5.2, and Subsection 7.1, respectively.

#### **4.3 Operational Information Examination and Verification**

OPG Safeguards Officer promptly facilitates access required for the CNSC or the IAEA to confirm the accuracy of operational information submitted annually as per Subsection 4.2. Access to such information is facilitated through an Unannounced Inspection, a Short Notice Random Inspection, or a CA. As a good industry practice, OPG updates the operational program when changes occur that are of relevance to safeguards, or on a quarterly basis confirming no changes. Where safeguards relevant changes occur, OPG notifies the CNSC of such changes.

## 5.0 NUCLEAR FUEL LIFE CYCLE

The BWRX-300 is an “off-load reactor” as defined in IAEA TRS No. 392 (Reference SG.8). Refueling operations, and associated fuel movements for the BWRX-300 are conducted using 12-24 months fueling cycles Chapter 1, Table 1.5-1, and with a large number of assemblies moved during refueling. Refueling operations occur with an open RPV and fuel transfers occur in transparent water shielding. This combination of off-load refueling and transparent water shielding permits direct visual observations for safeguards purposes.

The BWRX-300 integrated reactor building structure consists of the RB, containment, and containment internal structures. As described in Chapter 9B, Subsection 9B.2.3, and shown in Figure 9B-1, the cylindrical-shaped below-ground portion of the RB structure houses the primary reactor system, reactor support and safety systems, reactor containment, and most of the associated essential power supplies and equipment. The rectangular prism-shaped above-ground portion of the RB structure houses the refueling floor, refueling and fuel handling systems, and an overhead crane.

Among its primary functions, the RB houses and structurally supports the RPV, the containment vessel, the reactor support structure of the primary reactor system, fuel pool, fuel handling and storage equipment, biological shielding, and associated equipment and structures. The RB also provides protection for safeguards equipment from environmental and external hazards such as floods, winds, tornadoes, and earthquakes.

The BWRX-300, for the safeguards purposes, is a “Type I facility design” (Reference SG.8) since the fuel pool is within the RB. As such, the safeguards C/S measures (Reference SG.10) apply at all times to fresh fuel receipts, core fuel during refueling and irradiated fuel in the fuel pool.

As described in Chapter 4, Subsection 4.1.4, the reactor core arrangement contains enriched fuel assemblies as well as reactivity control rods. The coolant which acts also as moderator is light water that flows upward past the fuel rods. The core arrangement and coolant are contained within a heavy walled RPV, as shown in Chapter 4, Figure 4.1-1, and detailed in Chapter 5, Section 5.4. The BWRX-300 fuel assemblies are designed to maintain fuel cooling and to avoid fuel damage under normal operation and anticipated operational occurrences, as described in Chapter 4, Subsection 4.4.1.

The BWRX-300 reactor core comprises two hundred forty (240) fuel assemblies and fifty-seven (57) control rods with an N-lattice configuration (see Chapter 4, Figure 4.2-2) and is designed to utilize commercially available Boiling Water Reactor (BWR) fuel (refer to Chapters 4, Subsection 4.2.1 for fuel system design bases information, Subsection 4.2.2.2, for additional information on GNF2 fuel cladding, Subsection 4.3.5 for key features of the BWRX-300 reactor core, and Subsection 4.4.1 for steady-state operation). Information on the nuclear boiler system is found in Chapter 5, Subsection 5.3.1.

OPG has a fuel management plan that addresses the safety, reliability, and security implications, as well as allowances for operational procedures. The significance of the fuel management plan persists for the fuel life cycle, including long-term storage of irradiated fuel assemblies. Current OPG fuel management plan is to be updated or a new plan is to be developed with specific DNNP site and BWRX-300 nuclear fuel information to support OPG’s application for a “Licence to Operate”, as per Section 4.0 of CNSC REGDOC-1.1.3 (Reference SG.18).

As described in Subsection 2.1, BWRX-300 uses Group 1A nuclear material. Group 1 material is uranium, plutonium-239, and thorium that are sufficiently pure to be used for fuel fabrication or for isotopic enrichment. This includes all materials in the nuclear fuel cycle past the point of core concentrate, or anything created from such materials, including depleted uranium. As such, a fuel

management plan describing the flow and storage of nuclear material is a key aspect that is considered in the design and operation of BWRX-300 facility and is part of satisfying the requirements of CNSC REGDOC-2.13.1 Safeguards and Nuclear Material Accountancy (Reference SG.6).

The fuel handling and storage systems, described in Chapter 9A, Section 9A.1, are designed to address nuclear safety, security, and IAEA safeguards, as well as prevent inadvertent criticality and maintain shielding and cooling of irradiated fuel as necessary to meet operating and off-site dose constraints.

## **5.1 Flow and Storage of Nuclear Fuel**

Fabrication, movement, handling, and storage are important activities for accounting for the nuclear fuel. This subsection describes such activities as they apply to the BWRX-300 design.

### **5.1.1 Fabrication and Transport**

The BWRX-300 GNF2 fuel is manufactured by GNF. The GNF2 fuel has features used in previous GNF fuel designs including Zirconium-lined barrier cladding, high performance spacers, water rods, part length rods, thick corner/thin wall channel, and customized assembly nuclear designs.

Refer to Chapter 4, Subsection 4.2.2, for information regarding GNF2 fuel assemblies.

Each fuel assembly has a serial number (or handle ID) that is unique to each fuel assembly. The serial number provides for traceability of an assembly throughout the life cycle from manufacturing to long-term storage.

The fresh fuel assemblies are packaged by GNF in approved shipping containers and transported to OPG's DNNP site. These GNF and OPG activities are performed under separate provisions of import and export requirements, as well as governed by licenses to transport issued by relevant regulatory authorities in the United States and Canada.

The number of fresh fuel assemblies required to maintain reliable production for BWRX-300 is based on core nuclear design, which is described in Chapter 4, Subsection 4.3.2. The fresh fuel assemblies are shipped and stored on-site upon receipt in advance of the initial core fueling and in advance of refueling outages, prior to loading fuel into the reactor. Upon arrival of the fresh fuel shipment at the site entry-point gate, a receiving survey is performed by radiation protection and security. Once radiation protection receiving is completed, the shipment proceeds with radiation protection, security, safeguards, and operations oversight to a designated on-site truck receiving area. The shipment is then subject to inspection and temporary staging as described in Subsection 5.1.2.

### **5.1.2 Fresh Fuel Staging and Storage**

Fresh fuel staging and storage plans are developed and will be finalized prior to the first fresh fuel shipment. The planned process of handling and staging the BWRX-300 fresh fuel for inspection involves unloading the fresh fuel containers from the receiving truck through post shipment arrival inspections and ends when the fresh fuel assemblies are stored in the fuel pool racks for the initial core fueling or in advance of a refueling outage. The process described in the subsequent paragraphs is based on experience at existing BWR nuclear plants and is expected to be representative of the process that will be employed for the BWRX-300 facility.

Each fresh fuel shipment is accompanied by a document which includes all necessary information to meet the requirements of CNSC REGDOC-2.13.1 (Reference SG.6). Upon receipt inspection, the received nuclear material is added to OPG inventory by updating the nuclear material accountancy software (refer to Subsection 5.2).



NEDO-33972 REVISION 0  
NON-PROPRIETARY INFORMATION

Representative licensed fresh fuel shipping containers can hold a maximum of two BWRX-300 fuel assemblies. Each container has both an inner and an outer box. A receiving inspection is completed on the inside, and outside of the shipping containers. Inner boxes, each containing a maximum of two fuel assemblies, are removed, and stacked at the end of the receiving area. The outer boxes stay staged in the location waiting the empty inner boxes at the completion of the fuel receiving activities.

For a BWR nuclear plant, the fresh fuel is routinely staged in the fuel pool prior to the first fueling and subsequent refueling outages. Receiving fresh fuel typically follows a planned path of movement involving several steps for receiving, staging, inspection, and storing in the fuel pool. Such steps are generally applicable to the BWRX-300 planning for receipt of fresh fuel, as follows:

1. Inner boxes are moved through the RB equipment hatch and set in the RB.
2. Inner boxes are then lifted up to the refuel floor and staged in stacks in accordance with a box-stacking plan.
3. Moving fresh fuel to the fuel pool begins by selecting the first fresh fuel inner box, which is moved to a staging location on the refuel floor designated for one box.
4. That inner box lid is removed, the box is inspected to verify conditions, the box is up-righted using the overhead crane and placed in a dry up-righting inspection stand. Typically, up to two fuel assemblies are now vertical in the inner box with bail handles on top. Trained fuel inspectors prepare the two fresh fuel assemblies for inspection, inventory, and installation into the fuel pool.
5. Fresh fuel assemblies are moved into a location known as fuel channel storage within the fuel pool. The serial numbers of the channeled fresh fuel assemblies are checked against the tracking documents (which may be referred to as "move sheets") and then recorded using OPG's inventory sheet. Fresh fuel is then moved to the fuel pool for storage until the outage.

The fuel pool comes equipped with two fuel elevators, also known as fuel preparation machines that are attached to a wall of the fuel pool (refer to Chapter 9A, Subsection 9A.1.2.3). The primary use of the fuel preparation machine is in loading fresh fuel into specific storage pool locations prior to a refueling outage.

Once all fresh fuel inspection is completed, and the fresh fuel assemblies are loaded into the fuel pool, the shipping containers are moved and prepared for return shipment. This activity completes the receiving, and inventories of the fresh fuel. At this time, the locations of the new fresh fuel assemblies are logged in the OPG nuclear material accountancy software.

The BWRX-300 planned design lifecycle is 60 years during which fresh fuel arrives, and irradiated fuel is discharged from the reactor core to the fuel pool in fueling cycles in accordance with a fuel management plan that is developed specifically for the BWRX-300 facility.

Irradiated fuel assemblies are moved into casks for dry storage, as explained in Subsection 5.1.4.

The design and implementation of handling and storage of the fresh fuel, as described in Subsections 5.1.1 and 5.1.2, satisfy the requirements of Subsection 8.12.1 of CNSC REGDOC-2.5.2 (Reference SG.14).

Measures and verification activities related to nuclear material accountancy regarding receiving fresh fuel or shipping nuclear material off-site are detailed in Subsection 7.2.1.

### **5.1.3 Core Fuel Shuffling/Refueling**

The BWRX-300 RB design has a refueling machine for fuel movement and servicing. The refueling machine is a gantry crane, which spans the RPV and the fuel pool. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel assemblies for placement in the core or storage racks.

A position indicating system and travel limit computer locate the grapple over the RPV core and prevent collision with pool obstacles. The mast grapple has a redundant load path so that no single component failure results in a fuel assembly drop. Interlocks on the fueling machine:

1. Prevent hoisting a fuel assembly over the vessel unless an all-control-rod-in permissive is present
2. Limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit
3. Prevent lifting of fuel without grapple hook engagement and load engagement

These measures for core fuel shuffling and refueling satisfy the requirements of Subsection 8.12.1 of CNCS REGDOC-2.5.2 (Reference SG.14).

Upon completion of fuel movement operations, it is important that a core verification task is accomplished in which individual fuel assemblies are verified to be installed in the correct position, using serial numbers, consistent with the core reload analysis. At this time, the current and new locations of the nuclear fuel assemblies are updated/logged in the nuclear material accountancy software.

### **5.1.4 Irradiated Fuel Storage**

Irradiated fuel storage includes wet storage in the fuel pool as described in Subsection 5.1.4.1 and long-term dry storage as it becomes available as described in Subsection 5.1.4.5. The information in the following subsections is based on proven BWR designs and operating experience and, since the BWRX-300 is the tenth evolution of BWR nuclear plants, is applicable to the BWRX-300 for fuel storage planning. Factors such as pool cooling (refer to Chapter 9A, Subsection 9A.1.3.2), and rack spacing rules, are considered to establish, in a timely manner, a plan for cask campaigns to remove irradiated fuel from the fuel pool to short-term or long-term irradiated fuel storage facilities.

#### **5.1.4.1 Wet Storage**

The BWRX-300 utilizes a fuel pool for the interim storage, and cooling of the irradiated fuel. As described in Chapter 1, Subsection 1.5.2.1, the BWRX-300 fuel pool is located in the RB.

The fuel pool size is determined largely by the volume of cooling water needed for a 7-day coping period. Fuel preparation machines are located on the fuel pool periphery. Used reactor core instruments are also stored in the pool and cut up and packaged, as needed. There are no through-wall penetrations in the fuel pool, and the design of the fuel pool gate prevents leakage during any line-break accidents. Monitoring and controlling the water level in the fuel storage pool is discussed in Chapter 9A, Subsection 9A.1.3.

The racks in the fuel pool have sufficient capacity to accommodate wet storage of at least 8 years of operation plus one core load of fresh fuel and one full core off-load, as indicated in Chapter 1, Table 1.5-1. The BWRX-300 fuel pool racks have a storage capacity of approximately 250% of core capacity, which is equivalent to a minimum of 600 fuel assemblies.

The serial number on top of the bail handle, which is part of the upper tie plate of every fuel assembly, is used to specify each assembly's location in the core for each cycle. As the operator

of the BWRX-300, OPG strictly follows all special nuclear material accountancy rules, always retains records for every movement of an assembly over the assembly's life and updates the nuclear material accountancy software to reflect field conditions.

The Fuel Pool Cooling and Cleanup System (FPC) removes decay energy from the fuel pool (refer to Chapter 9A, Section 9A.1.3 for further information on Fuel Pool Cooling and Cleaning System).

A full array in the loaded fuel rack is designed to be subcritical by at least 5%  $\Delta k$  under all normal, and abnormal conditions, and safety analyses are performed using validated codes, as described in Chapter 9A, Subsection 9A.1.2.2.

The fuel pool is protected from malevolent acts, seismic activity, internal and external missiles, and loss of cooling due to anticipated operational occurrences. Relevant design and safety aspects are found in Chapter 3, Section 3.5, and Chapter 15, Sections 15.5 and 15.7.

The design and operational measures described in this subsection meet the requirements and guidance of Subsection 8.12.2 of CNSC REGDOC-2.5.2 (Reference SG.14) for handling and storage of irradiated fuel.

#### **5.1.4.2 Dry Storage**

As described in Chapter 9A, Subsection 9A.1.2.3.5, spent fuel is transferred from the fuel pool in a transfer cask using a single failure proof Polar Crane. Handling systems for fuel cask loading do not perform any Safety-Category function during normal or off-normal conditions. The system and equipment functions associated with the handling system for fuel cask loading, provide the means to transfer spent fuel located in the fuel pool into a spent fuel storage cask and transport of the spent fuel storage cask to a location for long-term storage. Refer to Chapter 9A, Subsection 9.1.4 for detailed information for demonstrating compliance to fuel cask loading requirements in CNSC REGDOC-2.5.2, Subsection 8.12.1.

If irradiated fuel is managed in another Material Balance Area, the nuclear material is removed from the nuclear material accountancy inventory following transfer to dry storage (Reference SG.16). The nuclear material is added to the Material Balance Area for the dry fuel storage by an Inventory Change Document. The process is completed by the submission of an Inventory Change Document to the CNSC and to the recipient who is responsible for the Material Balance Area for the dry fuel storage. If OPG has dry fuel storage within the same Material Balance Area, positions are updated in the nuclear material accountancy software, under new Key Measurement Point.

#### **5.1.4.3 Canister Loading**

Canister loading is described in Chapter 9A, Subsection 9A.1.4.3.2.

Some fuel canisters may contain damaged fuel assemblies depending on OPG's specific fuel loading schemes.

#### **5.1.4.4 Canister Movement to Dry Storage**

Canister movement from the fuel pool to an on-site dry storage installation (regulated under a separate licence) is performed under the guidance and participation of OPG Radiation Protection, Security, Safeguards and Health Physics. The RB Polar Crane is used to lift a transfer cask to the refuel floor and subsequently to the truck bay. Once the dry, sealed, and decontaminated canister is placed in the transfer cask, the transfer cask is picked up by the hauler and moved to the dry storage installation. The canister is transferred from the transfer cask to the storage cask on the dry storage pad.

#### **5.1.4.5 Long-Term Storage**

The Nuclear Waste Management Organization (NWMO) is a not-for-profit organization established in 2002 by Canada's nuclear electricity producers in accordance with the Nuclear Fuel Waste Act (Reference SG.11) and is responsible for designing and implementing Canada's plan for the safe, long-term management of used nuclear fuel. The plan, known as Adaptive Phased Management, requires used fuel to be contained and isolated in a deep geological repository. The plan also employs a comprehensive process to select a site with informed and willing hosts for the project. Current NWMO design is focused on receiving, transporting, preparing, and burying CANDU fuel assemblies which are significantly smaller than BWRX-300 fuel assemblies. Evaluation of NWMO potential design for dealing with the BWRX-300 irradiated fuel assemblies includes options such as, but is not limited to, the following:

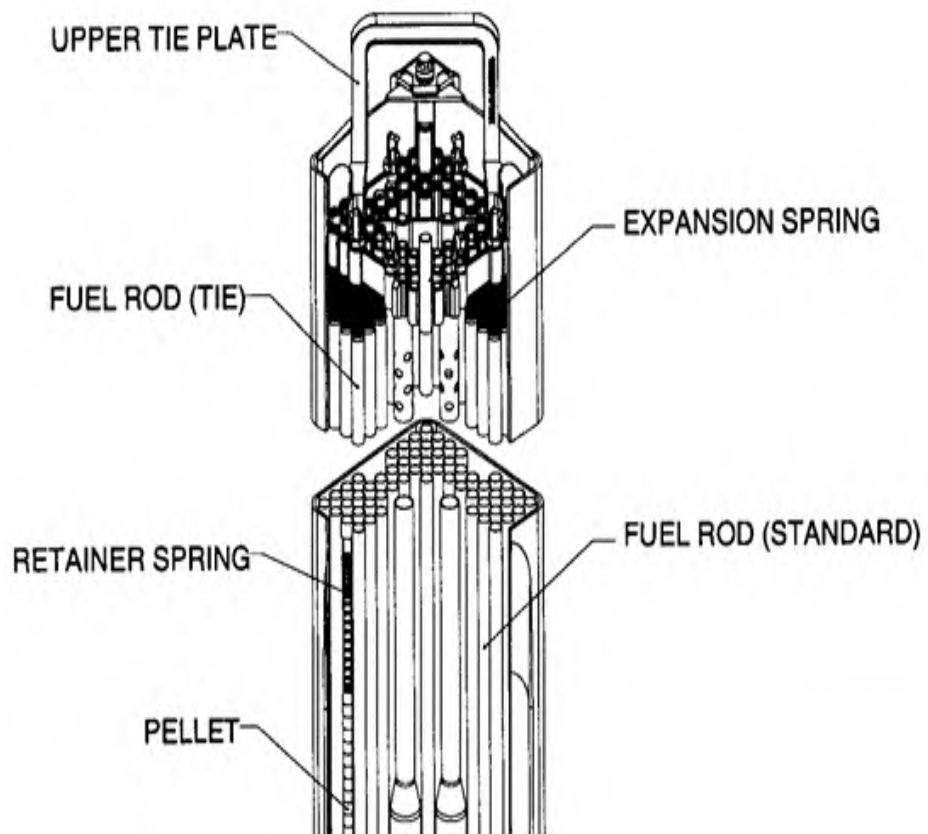
1. Using an elongated CANDU fuel canister, maintaining the diameter and materials of construction while lengthening the canister to accept BWRX-300 fuel, most likely three fuel assemblies
2. Using the canister currently used in Finland designed to accommodate 12 BWR fuel assemblies. This canister has a larger diameter, longer length, and different materials of construction
3. Cutting BWRX-300 fuel assemblies to fit inside the current proposed CANDU fuel canister
4. Developing a segmented fuel product line with the ability to fit the segments into a CANDU fuel canister

#### **5.2 Nuclear Material Accountancy Software**

A nuclear material accountancy Software, which is subject to OPG quality management described in Chapter 17, Subsection 17.2.1.1, is used by OPG to track the fuel from the manufacturing point to the ultimate long-term storage. Communication and coordination are established between the manufacturer of the fresh fuel and OPG, the owner-operator of the BWRX-300 facility. The coordination includes information on, for example:

- Nuclear material accountancy
- Fuel Flow
- Fuel loading analysis
- Fuel burnup
- Foreign obligations / origin obligations

Such information is recorded, stored, and secured against physical and electronic sabotage in accordance with established OPG management system, as described in Section 4.0 and Section 7.0.



**Figure SG-5.1.1-1: Top of Fuel Assembly**

## **6.0 ACCESS AND ASSISTANCE TO THE IAEA**

The Safeguards and Nuclear Material Accountancy Program of OPG outlines the requirements for assistance and access to the IAEA for inspection or maintenance activities at OPG facilities (refer to Section 6 of CNSC REGDOC-2.13.1 (Reference SG.6)). An OPG Safeguards Officer is responsible to facilitate access to the IAEA for all reasonable requests, with the support of affected work groups.

Such access requires OPG qualified security or radiological escort to accompany the IAEA inspectors while they are on OPG sites. OPG's radiation protection program outlines requirements for accessing OPG facilities, and provisions are in place for IAEA inspector access. Safeguards Officers are responsible for facilitating access via OPG's Facility Access and Working Rights (Radiological) procedures. Safeguards Officers ensure that IAEA inspectors obtain access and radiological protection through cooperation with impacted work groups. A security or radiological escort assists the IAEA in completing their inspection or maintenance activities.

### **6.1 Safeguards Equipment and Seals**

Section 5 of CNSC REGDOC-2.13.1 (Reference SG.6) requires OPG to implement safeguards measures, install safeguards equipment, and provide services to such equipment at licensed facilities, with consultation and agreement with the CNSC. In support of such activities, OPG's Safeguards, and Nuclear Material Accountancy Program (Reference SG.16) provides direction inhibiting OPG personnel from damaging or interfering with IAEA equipment or seals. Individuals who operate or work in safeguards equipment are trained to identify, and not manipulate IAEA equipment or seals.

IAEA equipment and seals are within the access controlled protected area of the BWRX-300 facility, meaning that only individuals with appropriate clearance and training can access areas with such equipment or seals.

### **Safeguards Containment and Surveillance Equipment**

The Safeguards and Nuclear Material Accountancy Program of OPG (Reference SG.16) requires that individuals who work on, around, or near IAEA C/S equipment are trained to identify and not manipulate any IAEA C/S equipment. Such equipment is within the protected area, requiring appropriate clearance (security clearance or with a security escort) and radiological protection. As such, these areas are strictly access controlled, ensuring that untrained individuals or individuals with gaps in safeguards knowledge are not able to manipulate or impact the operation of IAEA C/S equipment. OPG develops operating procedures that outline required responses and notifications in the event that services are lost, or damage occurs to IAEA C/S equipment.

### **6.2 International Atomic Energy Agency Access**

Direction is provided in OPG's Safeguards and Nuclear Material Accountancy Program (Reference SG.16) for Safeguards Officers and Operations and Maintenance organizations to facilitate requests for access from the IAEA in a timely fashion (refer to Section 6 of CNSC REGDOC-2.13.1 (Reference SG.6) for relevant requirements). Requests involving unannounced inspections, short notice random inspections, and CA prompt OPG to provide prompt access to facilities and locations outside facilities under such requests. A longer period of planning is required for maintenance of safeguards equipment to ensure IAEA inspectors or maintainers are qualified to perform work under various regulations.

### **Inspection and Verification**

As detailed in Section 3.0, design information is provided to the IAEA via the DIQ, which is established for the pre-construction design of the facility. During construction, any changes that would impact the DIQ are communicated via the CNSC to the IAEA in a DIQ revision. During construction, the IAEA may arrange field activities to verify that the details in the DIQ match the on-site construction.

### **Complementary Access**

Subsection 6.3 of CNSC REGDOC-2.13.1 (Reference SG.6), requires that when a request for CA is received by OPG, prompt access, within 24 hours, is facilitated by an OPG Safeguards Officer, unless specified otherwise on the notice or within 2 hours if the inspector is already on site for another inspection.

### **Access to IAEA Equipment**

OPG is required to provide IAEA inspectors access, for all reasonable requests on short notice, to IAEA equipment following notification and identification of scope of activities. Where required, OPG facilitates the provision of ladders, scaffolding, lifting, or training. (Refer to Subsection 6.4 of CNSC REGDOC-2.13.1 (Reference SG.6)).

## **7.0 NUCLEAR MATERIAL ACCOUNTANCY**

Nuclear material and accountancy, including reporting, is driven by OPG's Safeguards and Nuclear Material Accountancy Program (Reference SG.16) – a program that applies at a high level to the DNNP, albeit the implementation means are tailored to the BWRX-300 facility.

### **7.1 Measurement of Nuclear Material**

The IAEA equipment is maintained by the IAEA on their own schedule, as required.

#### **Electronic Inventory Tracking**

The nuclear material accountancy software of OPG produces up-to-date List of Inventory Items that covers Group 1, 1A and 1B material in the BWRX-300 facility. The List of Inventory Items is submitted annually to the CNSC within seven business days after the physical inventory is completed, or at the request of the CNSC.

This nuclear material accountancy software also produces various inventory information such as:

- Key Physical Measuring Points Inventory Summaries
- General Ledgers, which show inventory changes including nuclear gains and losses
- Inventory Change Documents for receipt or shipment of nuclear materials

#### **Physical Inventory Taking**

A Physical Inventory Taking is conducted by OPG each calendar year and no greater than 14 months from the previous Physical Inventory Taking. Depending on operational considerations, the preferred Physical Inventory Taking date is reported by OPG in the operational program (refer to Subsection 4.2) that is submitted to the CNSC by September 30<sup>th</sup> of each year, which also outlines the activities of the subsequent year. During a Physical Inventory Taking, a review of the Group 1 material on the List of Inventory Items is conducted. The Safeguards Officer confirms the serial numbers and unique identifiers of the nuclear material along with their location against the List of Inventory Item. The Physical Inventory Taking is intended to verify that no Group 1 material is present that has not been included in the List of Inventory Items. Physical Inventory Takings are verified by the CNSC or IAEA inspectors under a CNSC or IAEA physical inventory evaluation or a physical inventory verification.

#### **Inventory Difference**

Although there is no expectation for a facility for safeguards purpose such as the BWRX-300 to have any material unaccounted for, if an inventory difference is discovered during a Physical Inventory Taking, OPG reports the inventory difference as required by Subsection 7.4 of CNSC REGDOC-2.13.1 (Reference SG.6).

#### **Foreign Obligations**

OPG verifies the serial numbers and unique identifiers of all material that is received by OPG at the facility. OPG also verifies the manufacturer information which outlines Canada's foreign obligations under foreign Nuclear Cooperation Agreements. Foreign materials under any nuclear cooperation agreement are tracked in a similar manner to that of all nuclear materials; that is, via the Nuclear Material Accountancy Program. This program identifies all foreign obligated materials at all Material Balance Areas and Key Measurement Points throughout the facility. Foreign obligated materials are reported annually via an Obligated Material Inventory Summary as required by Subsection 8.1.7 of CNSC REGDOC-2.13.1 (Reference SG.6).



## **7.2 Provision of Information**

The Safeguards and Nuclear Material Accountancy Program of OPG (Reference SG.16), along with the associated nuclear material accountancy software, facilitate the generation of required nuclear material accountancy reports for the BWRX-300 facility.

### **7.2.1 Nuclear Material Accountancy Reports**

OPG submits to the CNSC reports as outlined in CNSC REGDOC-2.13.1 Appendix D: Table D1 – Reporting Frequency For Nuclear Material Accountancy Reports. These electronic reports include the BWRX-300 facility data required by CNSC REGDOC-2.13.1 Appendix D: Table D2 – Labels and Data Elements Used in Nuclear Material Accountancy Reports (Reference SG.6).

#### **Electronic Reporting**

Subsection 8.1.1 of CNSC REGDOC-2.13.1 (Reference SG.6) requires OPG to report nuclear material accounting via CNSC's nuclear material accountancy reporting e-business system website. The nuclear material accountancy software generates these reports in the accepted formats.

#### **Inventory Change Document**

As required by Subsection 8.1.2 CNSC REGDOC-2.13.1 (Reference SG.6), OPG submits Inventory Change Documents for every increase or decrease, in terms of batches, of the nuclear material inventory. The nuclear material accountancy software generates an Inventory Change Document based on incoming information and verification upon receipt of new nuclear material, as well as upon shipment off-site of nuclear material to another location or a Material Balance Area.

Fresh fuel will have the tamper seals and serial numbers verified upon receipt. Shipment details from the vendor provide total amounts of nuclear material within the fresh fuel, including foreign obligated materials. The shipment information, including date, time, and amounts of nuclear material is uploaded to the Nuclear Material Accountancy Software, and the total amount of nuclear material accounted for increases accordingly. The nuclear material accountancy software generates a receipt Inventory Change Document in a predetermined CNSC and IAEA format. All fields of the Inventory Change Document are completed using codes defined in CNSC REGDOC-2.13.1 Appendix D: Table D3 – Inventory Change Codes

For shipments off-site, the nuclear material accountancy software generates the shipper Inventory Change Documents and provides necessary details for the receiving Material Balance Area to create a receipt Inventory Change Document. When nuclear material is removed from site, it is deducted from the total nuclear inventory for the Key Measurement Points for the Material Balance Areas. All Inventory Change Documents are submitted to the CNSC and IAEA via nuclear material accountancy reporting within one business day of the inventory change.

#### **General Ledger**

OPG Safeguards Officer submits, monthly, a General Ledger outlining all inventory changes that occurred in the BWRX-300 facility reporting month, including opening, and closing balance of nuclear material - as required by Subsection 8.1.3 of CNSC REGDOC-2.13.1 (Reference SG.6). The nuclear material accountancy software generates the General Ledger and manages via operator and Safeguards Officer inputting the amounts of nuclear material moved in or out of the Material Balance Area.

All transfers of nuclear material moved into or out of the Material Balance Area require an Inventory Change Document generated by the nuclear material accountancy software. The General Ledger is generated and reviewed by an OPG Safeguards Officer against all Inventory

Change Document information from the reporting period to verify the accuracy of the report. Once the accuracy of the report is verified it is then submitted to CNSC and IAEA via nuclear material accountancy reporting.

### **List of Inventory Items**

OPG submits a List of Inventory Items on an annual basis following the completion of a Physical Inventory Taking (refer to Subsection 8.1.4 of CNSC REGDOC-2.13.1 (Reference SG.6)).

Group 1A material List of Inventory Items is generated, reviewed, verified, and submitted after a Physical Inventory Taking is completed annually. Group 1B List of Inventory Items is generated and provided upon request from the CNSC.

### **Physical-Key Measurement Point Inventory Summary**

It is required, per Subsection 8.1.5 of CNSC REGDOC-2.13.1 (Reference SG.6), that OPG generates and submits to the CNSC and the IAEA a Physical-Key Measurement Point Inventory Summary that abridges the List of Inventory Item by grouping items with the same material codes into batches. The Physical-Key Measurement Point Inventory Summary is an output by the nuclear material accountancy software in an .xml format, to meet the schema outlined by the CNSC and IAEA.

### **Reconciliation Statement**

Annually, OPG submits to the CNSC a reconciliation statement following a Physical Inventory Taking of Group 1A material, as required by Subsection 8.1.6 of CNSC REGDOC-2.13.1 (Reference SG.6). The reconciliation statement calculates the inventory difference between the book ending inventory derived from the General Ledgers and Inventory Change Documents, and the physical ending inventory derived from the Physical-Key Measurement Point Inventory Summary or the List of Inventory Item.

### **Obligated Material Inventory Summary**

An Obligated Material Inventory Summary is generated before January 31 by the nuclear material accountancy software and is reviewed/verified by an OPG Safeguards Officer to ensure accuracy, and then submitted to the CNSC via the nuclear material accountancy reporting e-business. (Refer to Subsection 7.5 and Subsection 8.1.7 of CNSC REGDOC-2.13.1 (Reference SG.6).)

## **7.2.2 Design Information**

Section 3.0 describes in detail the design information that is provided to the CNSC via the submission of a DIQ. The DIQ is updated for reasons such as:

- Changes in field conditions
- Changes in essential information or parameters
- Requests for clarification from the CNSC or IAEA

## **7.2.3 Operational Information**

Operational information is provided to the CNSC annually by September 30<sup>th</sup> via the completion of the operational program (refer to Subsection 4.2). This operational program is also updated and provided to the CNSC once per quarter by the 15<sup>th</sup> of March, June, September, and December.

#### **7.2.4 Information Required by the Additional Protocol**

Subsection 8.4 of CNSC REGDOC-2.13.1 (Reference SG.6) requires OPG to submit a report to the CNSC by March 15<sup>th</sup> of each year with Canada's update as per the terms of the Additional Protocol (Reference SG.3). This includes the following information:

1. A general description of, and information specifying, the location of nuclear fuel cycle-related research and development activities, as defined in Appendix B: List of Nuclear Fuel Cycle-Related Research and Development Activities.
2. Current drawings of the site, a general description of each building on the site, including its use and, if not apparent from that description, its contents. Note that current site drawings need only be submitted if previously submitted site drawings are no longer accurate.
3. A description of the scale of operations for each location engaged in the activities specified in CNSC REGDOC-2.13.1 Appendix A: List of Declarable Nuclear-Related Manufacturing Activities.
4. Information regarding Group 1B material, as follows:
  - a. For materials exempted from safeguards pursuant to Article 37 of the Safeguards Agreement, the quantities, uses, and locations of such material
  - b. For materials exempted from safeguards pursuant to Article 36 of the Safeguards Agreement, but where the material is not yet in a non-nuclear end-use form, information regarding the quantities and uses at each location
5. Information regarding the location or further processing of intermediate or high-level waste containing plutonium, high enriched uranium-235 or uranium-233 on which safeguards have been terminated pursuant to Article 11 of the Safeguards Agreement.
6. General plans for the succeeding 10-year period relevant to the development of the nuclear fuel cycle (including planned nuclear fuel cycle-related research and development activities) when approved by the appropriate authorities in Canada.

#### **7.3 Retention of Records**

OPG Safeguards and Nuclear Material Accountancy Program (Reference SG.16) specifies the requirements for retention of records, as required by Section 9 of CNSC REGDOC-2.13.1 (Reference SG.6). Safeguards Officers maintain records in digital format on site specific shared drives. Records relevant for safeguards information is stored for the life of the facility.

## **8.0 SAFETY, SAFEGUARDS AND SECURITY INTERFACES**

In accordance with the requirements of Subsection 4.3.4 of CNSC REGDOC-2.5.2 (Reference SG.14), safety and nuclear security measures for the purposes of BWR-300 safeguards aspects are designed and implemented in an integrated manner as follows:

1. The BWRX-300 facility layout and site characteristics (refer to Chapter 2, Subsection 2.1.2) consider for the purpose of safeguards certain design requirements in an optimized configuration, such as:
  - a. Access routes for normal operational activities
  - b. Access control to minimize radiation exposures
  - c. Actions taken in response to internal or external hazards and operational events
  - d. Egress routes
  - e. Movement of hazardous substances, nuclear materials, and radioactive materials
  - f. Movement of authorized and unauthorized personnel
  - g. Interaction of building and support activities
2. Civil structures design, reactor design, safety analysis, and fuel handling and storage design taking into consideration safeguards requirements such as accessibility to, and effectiveness of safeguards equipment. (Refer to Chapters 3, Section 3.5 for general design aspects for civil engineering works of seismic category buildings and civil engineering structures; Chapter 4, Section 4.1 regarding a summary description of the RPV; Chapter 5, Section 5.1 concerning a summary description of the nuclear boiler system and associated systems; Chapter 15, Section 15.5 regarding deterministic safety analysis, and Chapter 9A, Section 9A.1 respecting information on fuel storage and handling systems.)
3. Security of safeguards information on nuclear material is governed by GEH and OPG procedures and is contained as part of a security annex to the Preliminary Safety Analysis Report. Cyber security is also included as part of that security annex. The information in a security annex is prescribed.

OPG procedures for the protection of safeguards information are based on the General Nuclear Safety and Control Regulations (Reference SG.5), Section 21(1)(a) and (d) for prescribed information, and Sections 23(1)(a)(iii) and 23(2) for transfer and disclosure.

## **9.0 SAFEGUARDS, AND IMPORT AND EXPORT**

Measures are established by OPG to proceed with:

1. The import of the un-irradiated BWRX-300 GNF2 fuel from the United States to Canada, in accordance with CNSC REGDOC-2.13.2 Safeguards and Non-proliferation - Import and Export (Reference SG.12) and the Nuclear Non-proliferation Import and Export Control Regulations (Reference SG.13).
2. An application for a separate CNSC licence to possess, transport, transfer, use, and store the GNF2 fuel in accordance with the requirements of the NSCA (Reference SG.4), and the General Nuclear Safety and Control Regulations (Reference SG.5).

## 10.0 REFERENCES

- SG.1 United Nations, "Treaty on the Non-Proliferation of Nuclear Weapons (NPT)."
- SG.2 IAEA INFCIRC/164, "Agreement Between the Government of Canada and the International Atomic Energy Agency for the Application of Safeguards in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," International Atomic Energy Agency.
- SG.3 IAEA INFCIRC/164/Add.1, "Protocol Additional to the Agreement Between Canada and the International Atomic Energy Agency for the Application of Safeguards in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," International Atomic Energy Agency.
- SG.4 Government of Canada, "Nuclear Safety and Control Act (S.C. 1997, c. 9)."
- SG.5 Government of Canada SOR/2000-202, "General Nuclear Safety and Control Regulations."
- SG.6 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."
- SG.7 CNSC Regulatory Document REGDOC-1.1.5, "Reactor Facilities: Supplemental Information for Small Modular Reactor Proponents."
- SG.8 IAEA Technical Reports Series No. 392, "Design Measures to Facilitate Implementation of Safeguards at Future Water Cooled Nuclear Power Plants," International Atomic Energy Agency.
- SG.9 International Atomic Energy Agency, "Safeguarding the Nuclear Future: Small Modular Reactors," <https://www.iaea.org/newscenter/news/safeguarding-the-nuclear-future-small-modular-reactors>
- SG.10 IAEA Safeguards Glossary, "International Nuclear Verification Series No. 3," International Atomic Energy Agency.
- SG.11 Government of Canada, "Nuclear Fuel Waste Act (S.C. 2002, c. 23)."
- SG.12 CNSC Regulatory Document REGDOC-2.13.2, "Safeguards and Non-proliferation - Import and Export."
- SG.13 Government of Canada SOR/2000-210, "Nuclear Non-proliferation Import and Export Control Regulations".
- SG.14 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- SG.15 CNSC Regulatory Document REGDOC-1.1.2, "License Application Guide: License to Construct a Reactor Facility."
- SG.16 N-PROG-RA-0015, "Safeguards and Nuclear Material Accountancy," Ontario Power Generation.
- SG.17 NEDO-33270, GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Global Nuclear Fuel - Americas, LLC.
- SG.18 CNSC Regulatory Document REGDOC-1.1.3, "Reactor Facilities – Licence Application Guide: Licence to Operate a Nuclear Power Plant."