

# REQUEST FOR CONFIDENTIALITY

## OF MATERIAL SUBMITTED IN RELATION TO NK054-CORR-00531-10740

### IMPORTANT NOTE:

The purpose of the confidentiality request process is to seek a decision from the Commission as to whether specific information being presented to a Commission proceeding can be protected. Generally, material received as part of a matter before the Commission is made available to the public by default. The rule of confidentiality (i.e., Section 12 of the [CNSC Rules of Procedure](#)) is applied only if the Commission decides in favour of a request for confidentiality.

Restricted access to proceedings and related material is exceptional, proportional, and minimal, and is not imposed lightly. Therefore, and to minimize the possibility of a challenge to a confidentiality ruling, the Commission weighs any request for confidentiality against the criteria set out in Section 12 to confirm that:

- the importance of protecting the information outweighs the public interest in public hearings and disclosure of evidence; and
- the confidentiality measures would affect the public nature of the proceeding only to the extent necessary to adequately protect the given information.

In the interest of enabling a timely decision, any request for confidentiality must be accompanied by redacted versions of all documents named in the request, and/or adequately informative summaries that can be made available to participants and the public. **Please provide the appropriate versions, as applicable.**

It is the responsibility of the person making the request to provide an adequately detailed explanation as to how and why subrule 12(1) applies.

In the matter of:

### OPG Application for a Licence to Construct a Single BWRX-300 Reactor at the DNNP Site

With regard to OPG Confidential Security-Protected submission - NK054-CORR-00531-10740 – Darlington New Nuclear Project – Submission of Package #3 Security Deliverables in Support of the Licence to Construct Application for the CNSC Review.

This request has been prepared in Canada, in the province of Ontario, in the matter of **OPG Application for a Licence to Construct a Single BWRX-300 Reactor at the DNNP Site**, scheduled for consideration in a Public Hearing, scheduled for October 2024 and January 2025.

I, **Andy Owen**, of **889 Brock Road, Pickering, Ontario, L1W 3J2**, am an authorized representative of the Ontario Power Generation Inc. I understand that:

- documents and information (“the material”) provided to the Canadian Nuclear Safety Commission (“the Commission”) as part of a public proceeding may be made publicly available;
- the material is considered confidential only if it is prescribed information under the [Nuclear Safety and Control Act](#) (NSCA), as defined in section 21 of the [General Nuclear Safety and Control Regulations](#), or if the Commission takes measures to protect the information; and
- regardless of any request for confidentiality or approval of same, the material may be disclosed if the Commission is required by law to disclose it (for example, after a request under [Access to Information Act](#)).

I hereby request that the Commission take measures to protect the following information, pursuant to rule 12 of the [Canadian Nuclear Safety Commission Rules of Procedure](#):

*Note: Where the request for confidentiality applies only to part of the submission, the portions to be deemed confidential must be clearly identified to distinguish them from any content that is non-sensitive.*

TABLE 1: MATERIAL TO BE DEEMED CONFIDENTIAL			
	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
<b>SCA: Security</b>			
1	<b>NK054-REP-00531-10000 – Construction Site Threat and Risk Assessment – New Nuclear at Darlington (R003)</b>	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential.  <b>**OPG's written PSAR is considered to provide a sufficient publicly-accessible summary of the material presented in this document.**</b>
2	<b>NK054-REP-61400-00001 Preliminary Safety Analysis Report (PSAR) Security Annex: Darlington BWRX-300 Security Assessment (R000)</b>	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential.  <b>**OPG's written PSAR is considered to provide a sufficient publicly-accessible summary of the material presented in this document.**</b>

**Detailed reason(s) for request:**

- The above-noted material should be protected for the following reasons:
  - These documents contain the detailed security plans, assessments and design parameters for the BWRX-300 facility in consideration for construction, that all fall under the definition of Prescribed Information..
- I attest that the above-noted material is not available through any public sources.

3. **MANDATORY:** I have included a **summary** or **redacted** version of the material that provides adequate detail to satisfy the public interest in public hearings and disclosure of evidence.
4. I understand that if this request is not approved by the Commission, I may withdraw the associated material within five business days of receiving written notice of the Commission's decision from the Commission Registrar (except as noted in items 5 and 6, below).
5. Notwithstanding item 4, above, I understand that if submission of the material is required pursuant to reporting requirements under the [NSCA](#) or the regulations under the NSCA, or pursuant to a licence issued under the NSCA, or if the material is specifically requested by the Commission, it may **not** be withdrawn.
6. I understand that upon receipt of this request, the Commission Registrar will treat the material that is subject to this request as confidential unless and until the Commission makes a ruling to deny this request.

**Attachments:**

- NK054-SR-01210-00001 R001, Preliminary Safety Analysis Report (PSAR) chapters 1, 2, 3, 13 and 14.

**Authorized signature:**



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Andy Owen, VP Security and Emergency Services

2024/07/26

Date




**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

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**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**

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**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.

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**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

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<b>Acronym</b>	<b>Explanation</b>
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

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<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

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## **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 (DNFS), 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 DNFS. 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

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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

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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.

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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.

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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.

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**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	

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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.

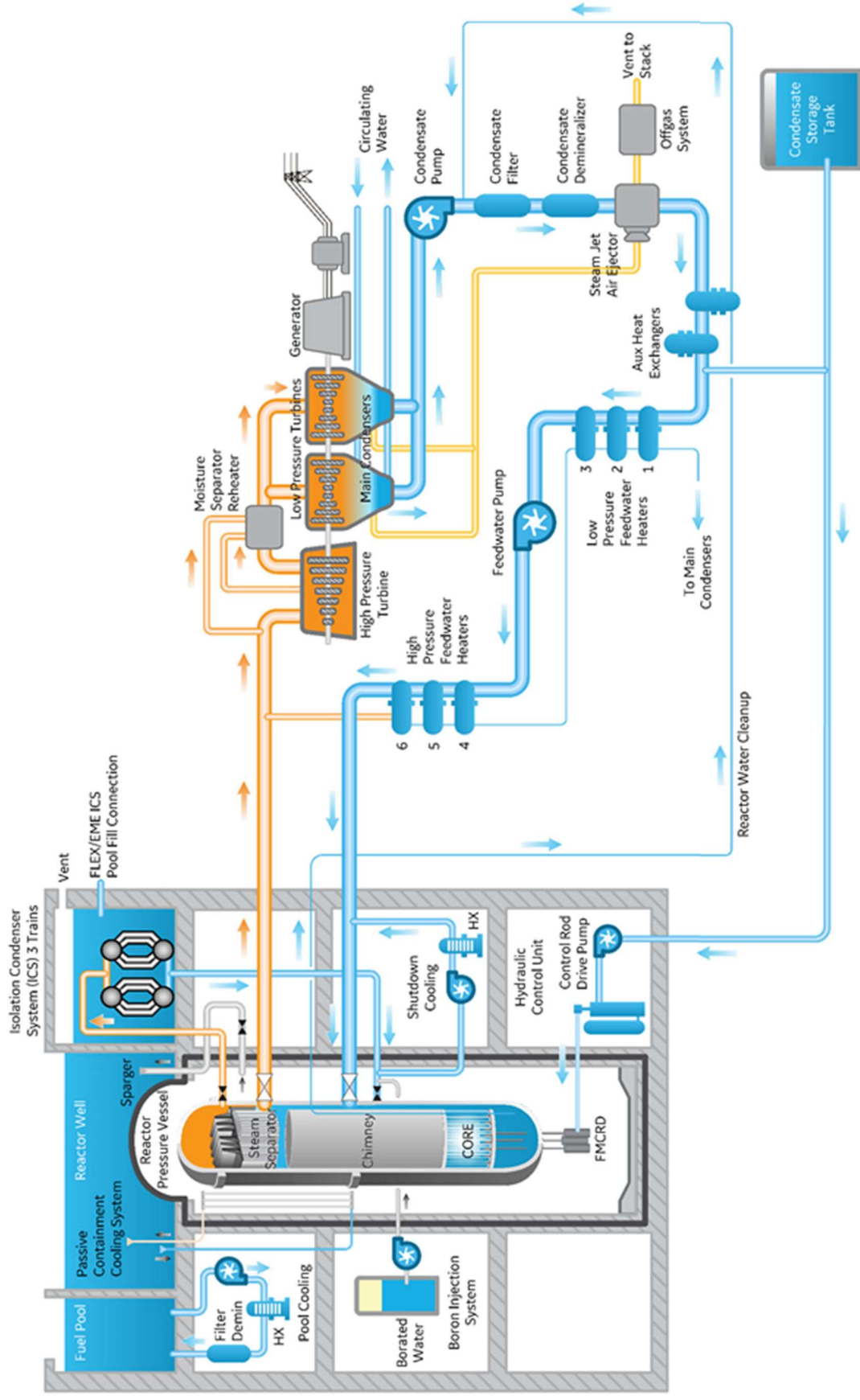


Figure 1.7-1: BWRX-300 Major Systems

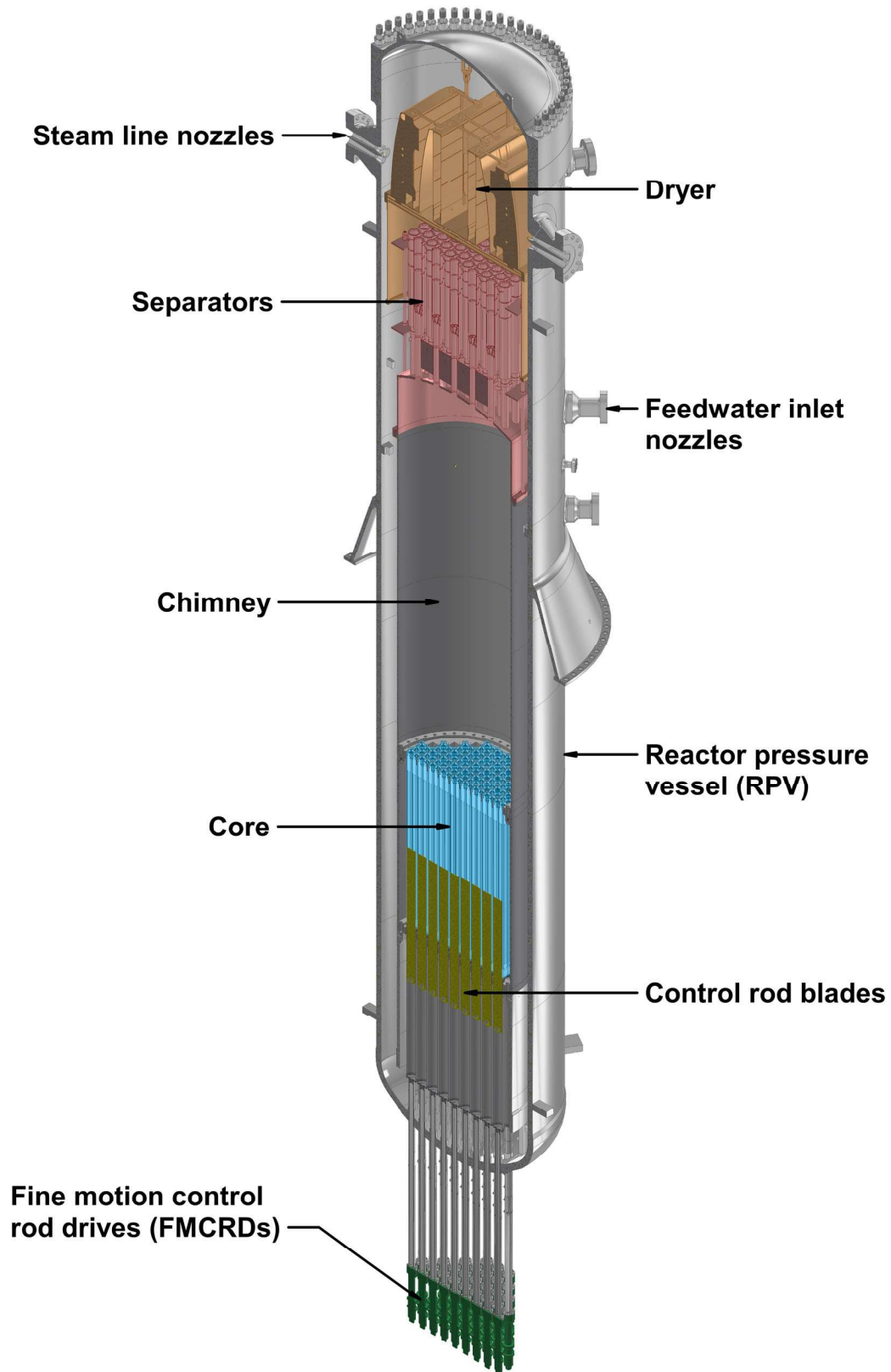
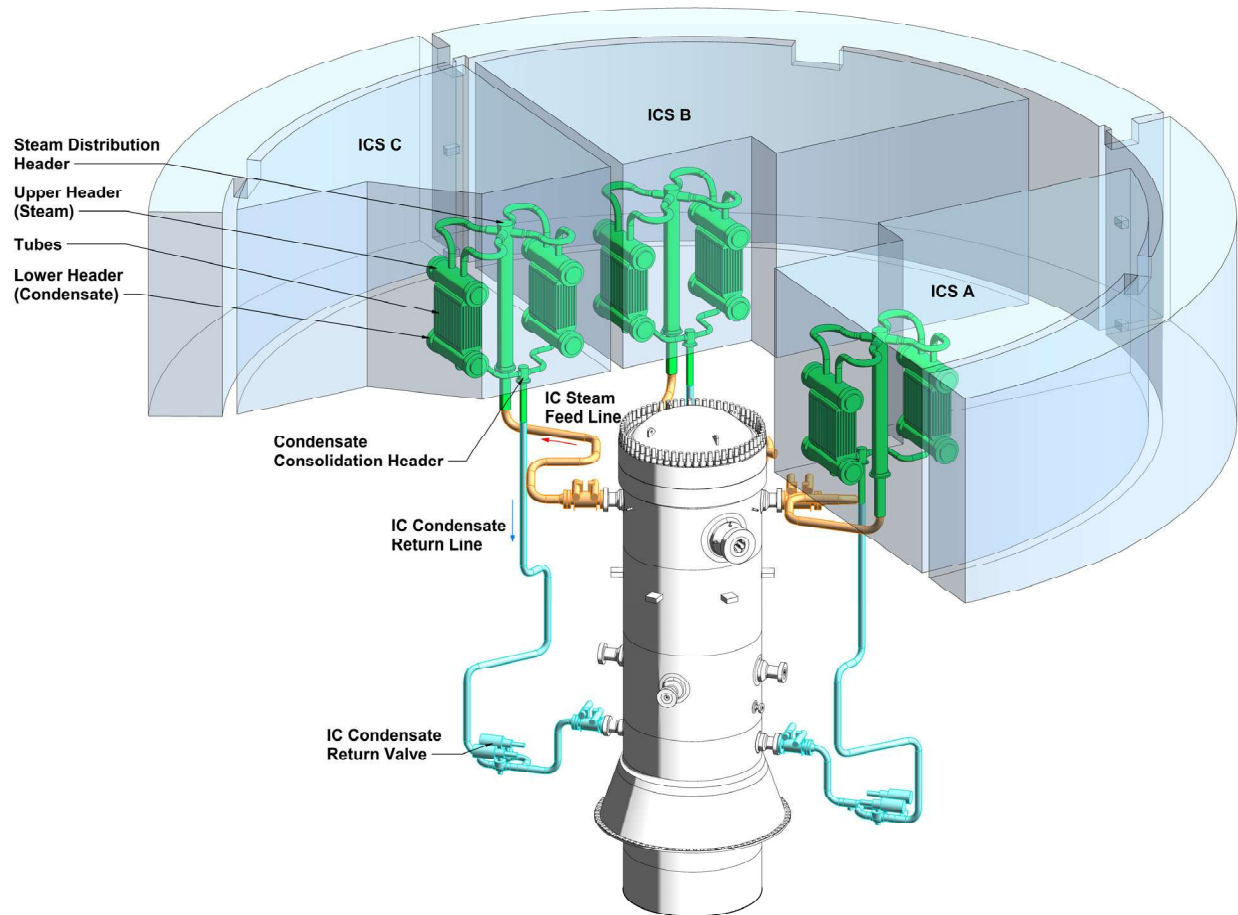


Figure 1.7-2: BWRX-300 RPV and Internals

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**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.

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**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

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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

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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**



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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.

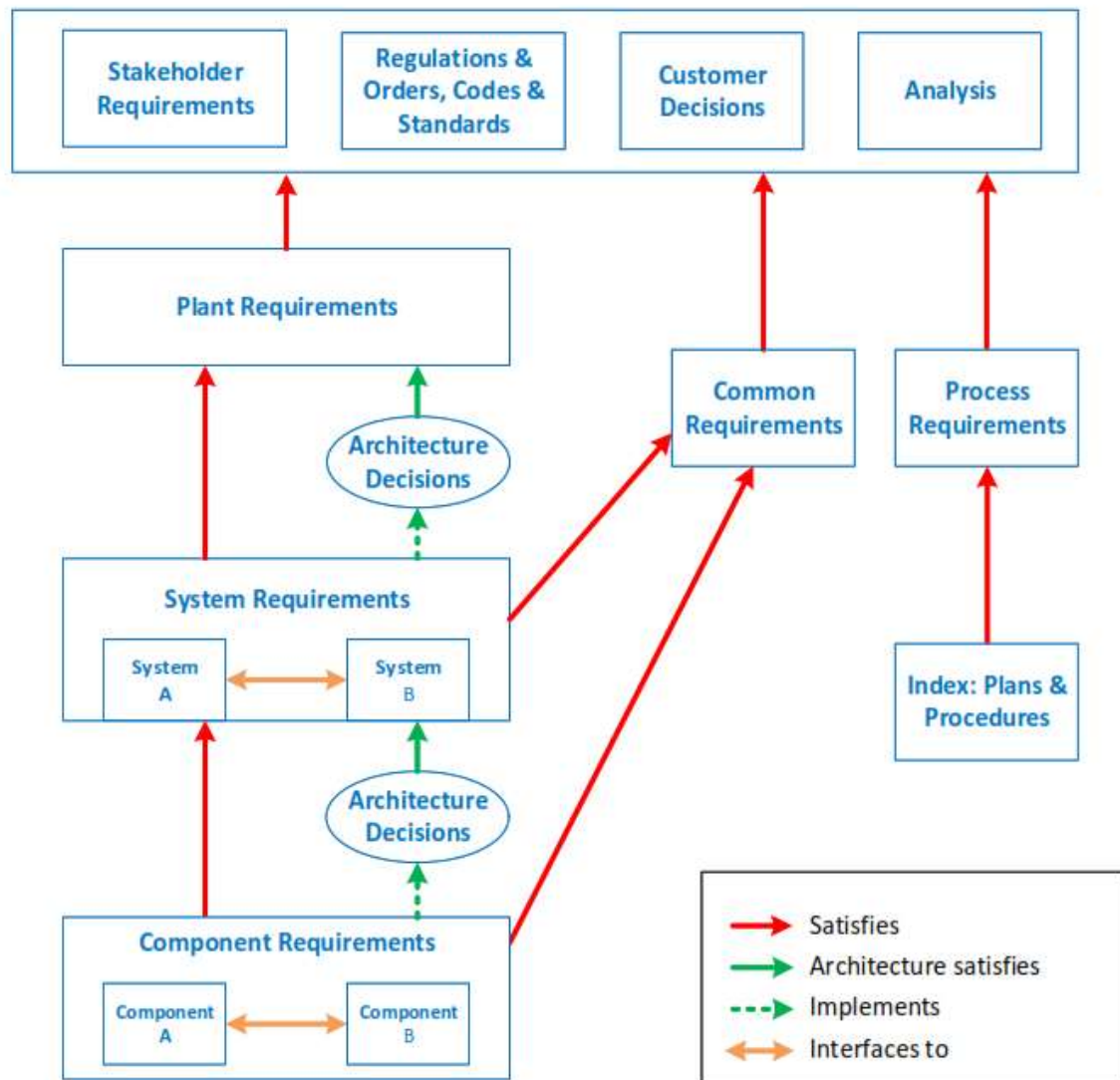


Figure 1.11-2: GEH Requirements Hierarchy

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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

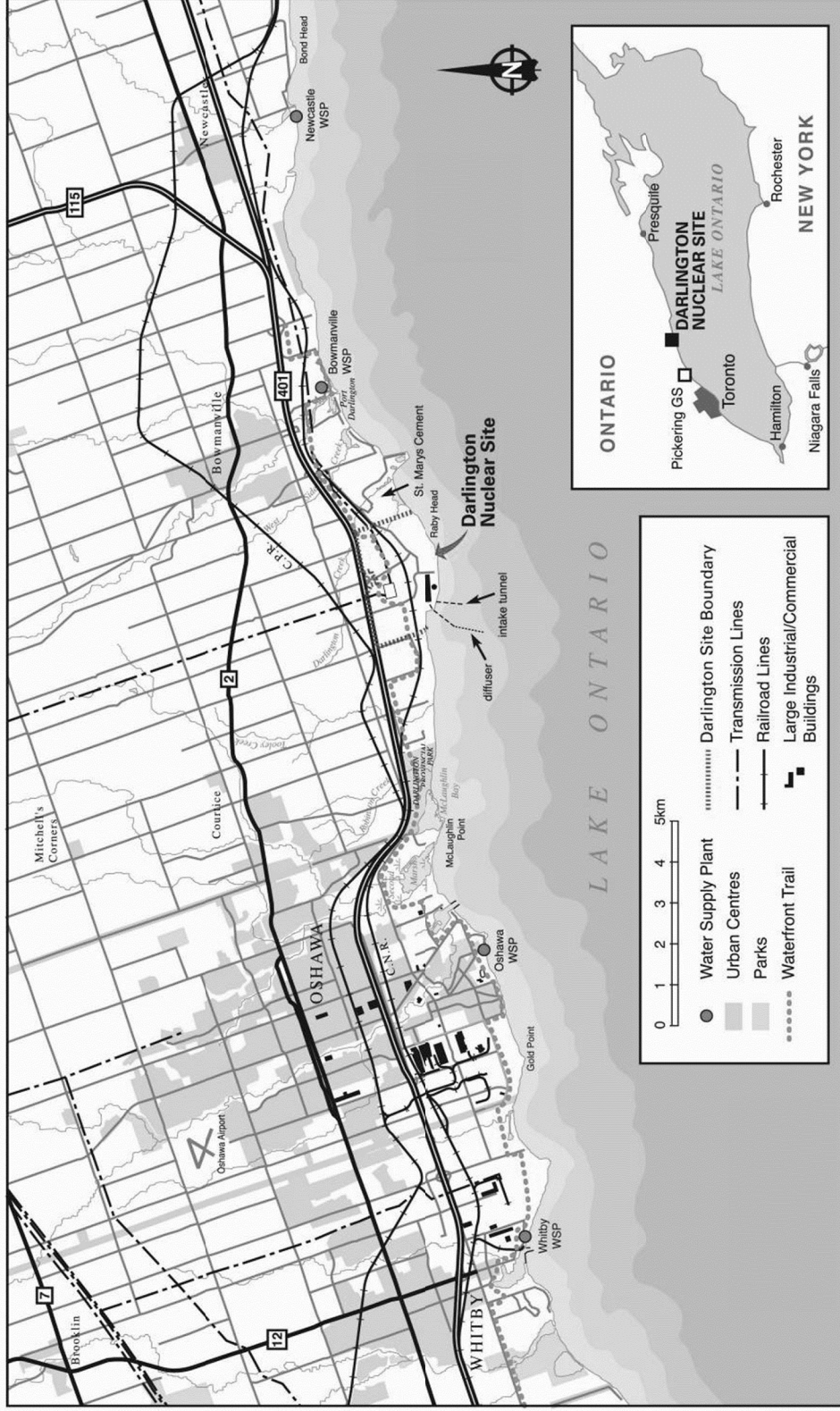


Figure A1.1-1: Darlington Nuclear Site Regional Location

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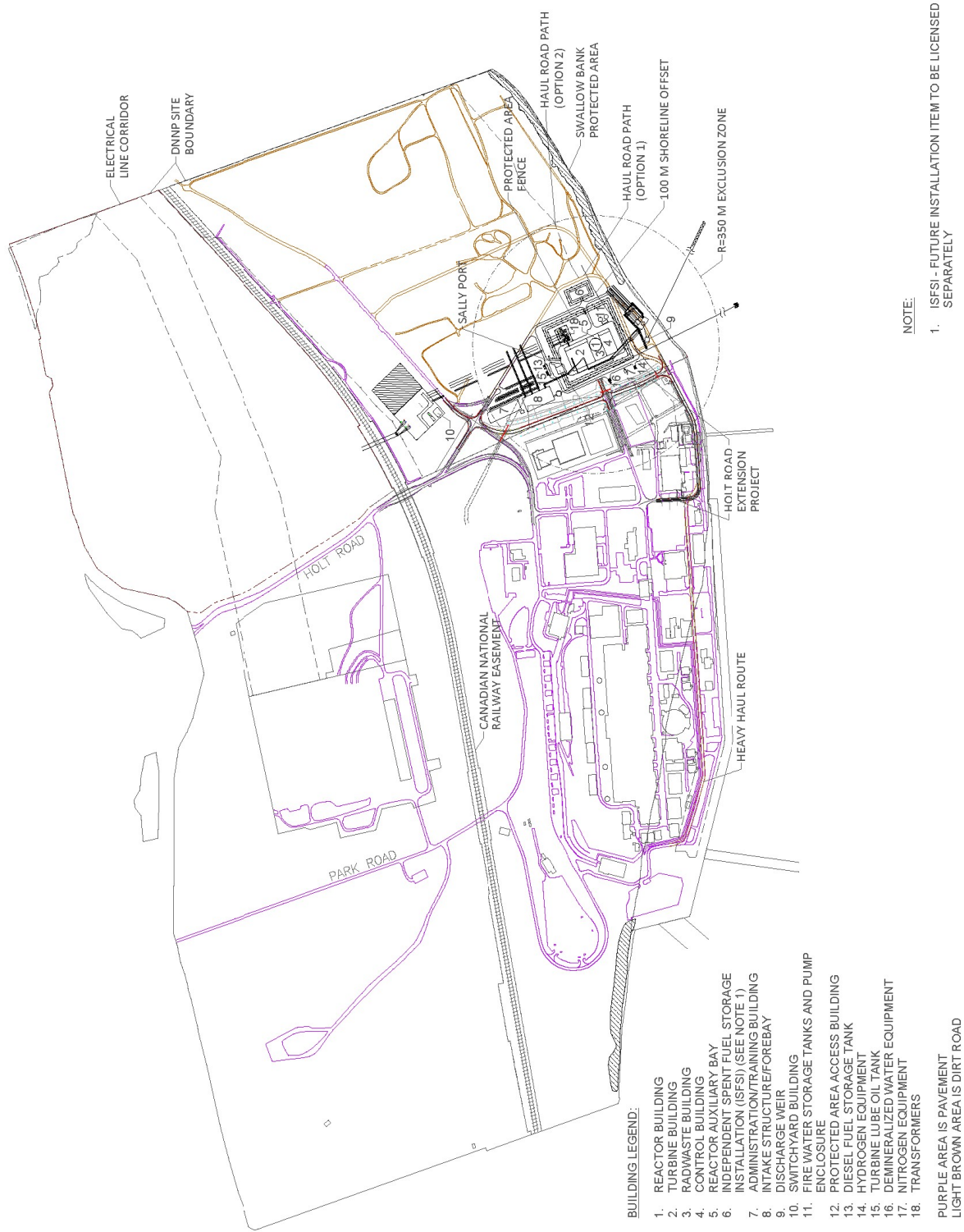


Figure A1.1-2: Darlington Nuclear Site (DNNP Proximity to DNGS)

**BUILDING LEGEND:**

1. REACTOR BUILDING
2. TURBINE BUILDING
3. RADWASTE BUILDING
4. CONTROL BUILDING
5. REACTOR AUXILIARY BAY
6. INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) (SEE NOTE 1)
7. ADMINISTRATION/TRAINING BUILDING
8. INTAKE STRUCTURE/FOREBAY
9. DISCHARGE WEIR
10. SWITCHYARD BUILDING
11. FIRE WATER STORAGE TANKS AND PUMP ENCLOSURE
12. PROTECTED AREA ACCESS BUILDING
13. DIESEL FUEL STORAGE TANK
14. HYDROGEN EQUIPMENT
15. TURBINE LUBE OIL TANK
16. DEMINERALIZED WATER EQUIPMENT
17. NITROGEN EQUIPMENT
18. TRANSFORMERS

**NOTE:**

1. ISFSI - FUTURE INSTALLATION ITEM TO BE LICENSED SEPARATELY

1-41



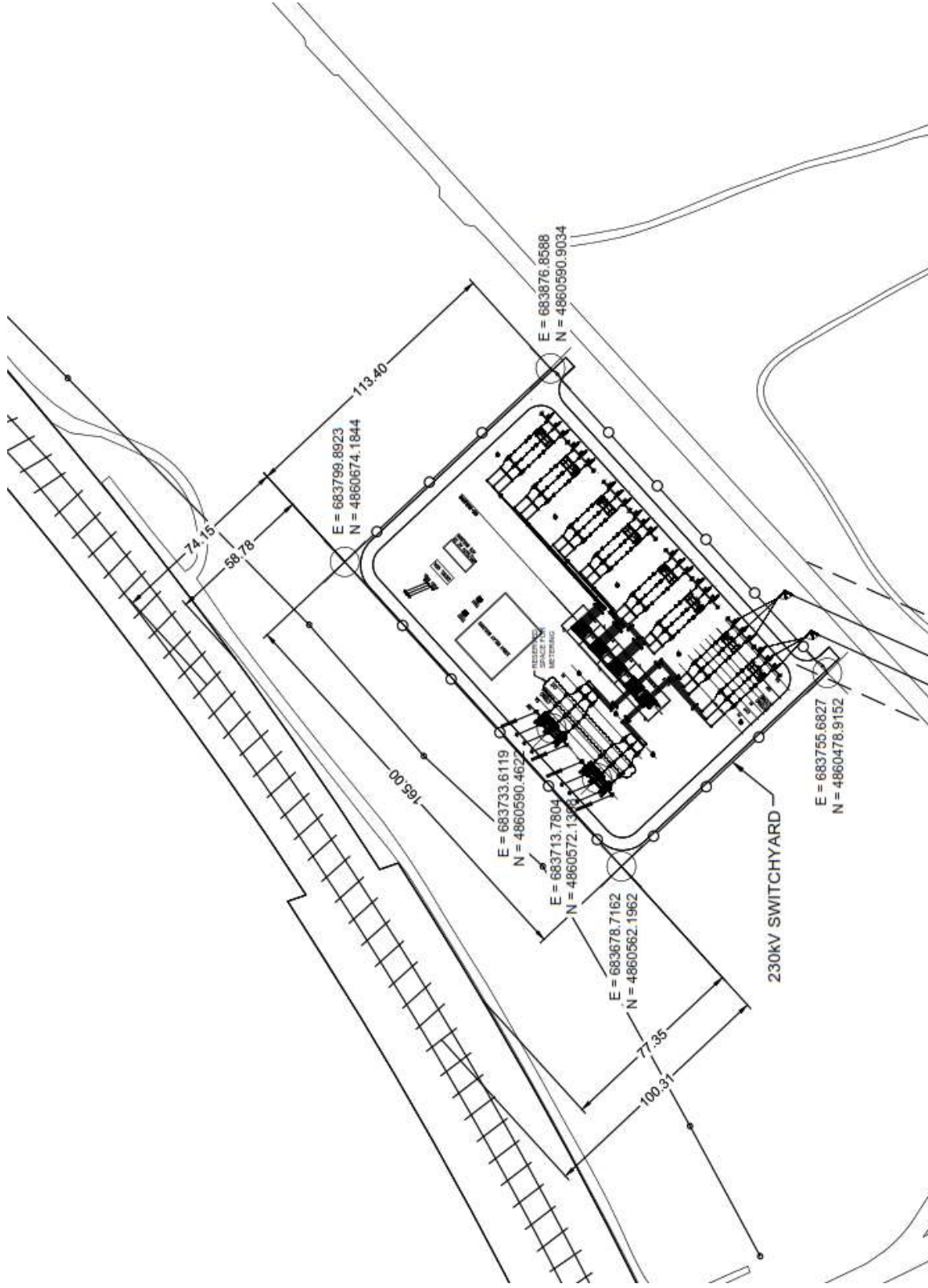


Figure A1.4-2: DNNP Switchyard Site Plan



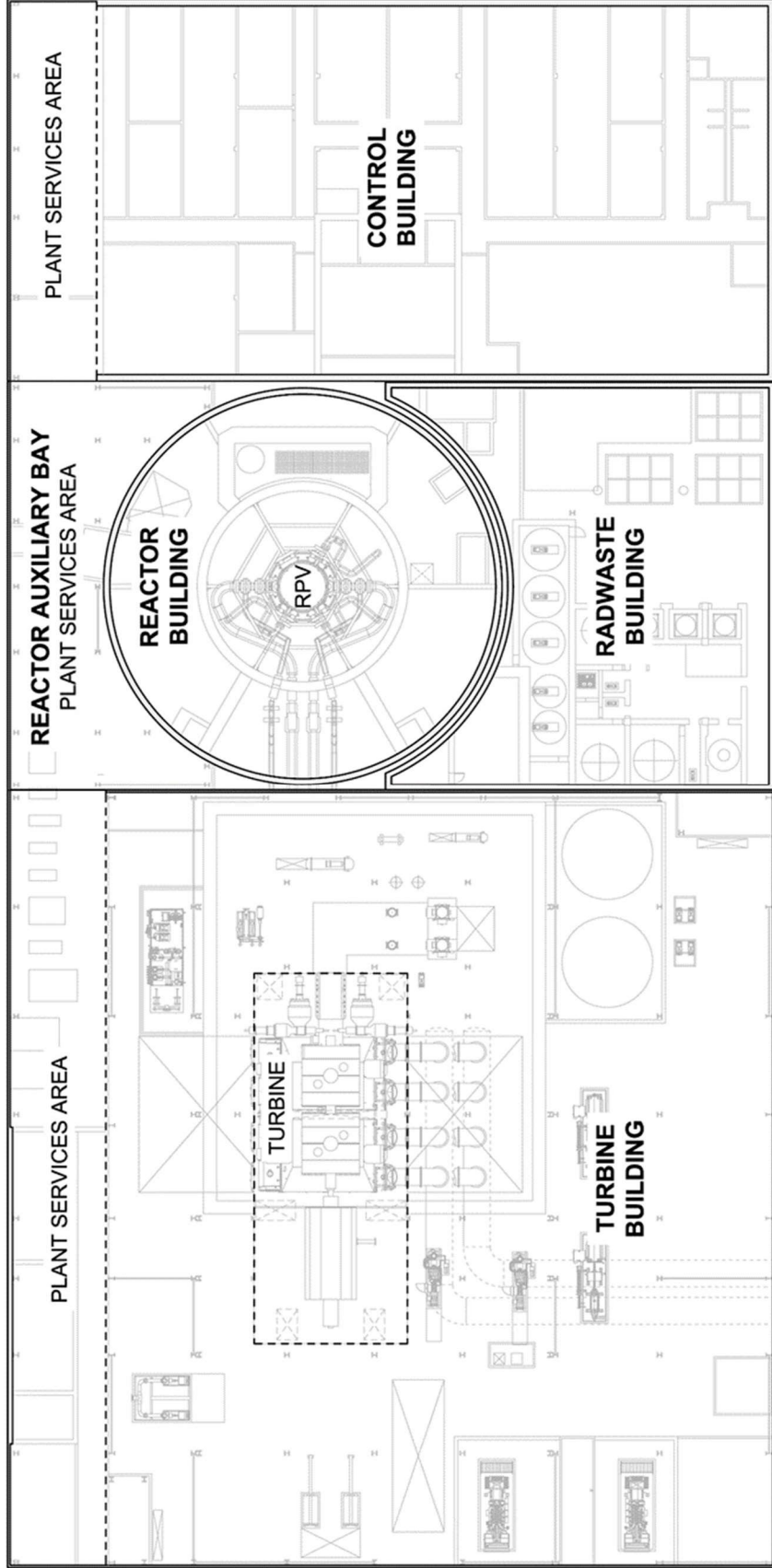


Figure A1.5-1: BWRX-300 Power Block Plan View at Elevation 0

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**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

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

<b>Document Number</b>	<b>Document Title</b>	<b>Doc Effective Date/Version</b>
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

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**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

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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

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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

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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).

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

<b>Document Number</b>	<b>Document Title</b>	<b>Doc Effective Date/Version</b>
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**

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**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

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<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

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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>

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		<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>

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		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>

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<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 BWXR-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

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<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

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<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
UHS	Uniform Hazard Response Spectrum
USNRC	United States Nuclear Regulatory Commission
WPCP	Water Pollution Control Plant

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## **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

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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

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- 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.

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**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

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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 (DN GS) 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 (DW MF), 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 DN GS 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 DN GS 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 DN GS 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 DN GS and includes:

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- 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

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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

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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.

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- 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.



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





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



2-18

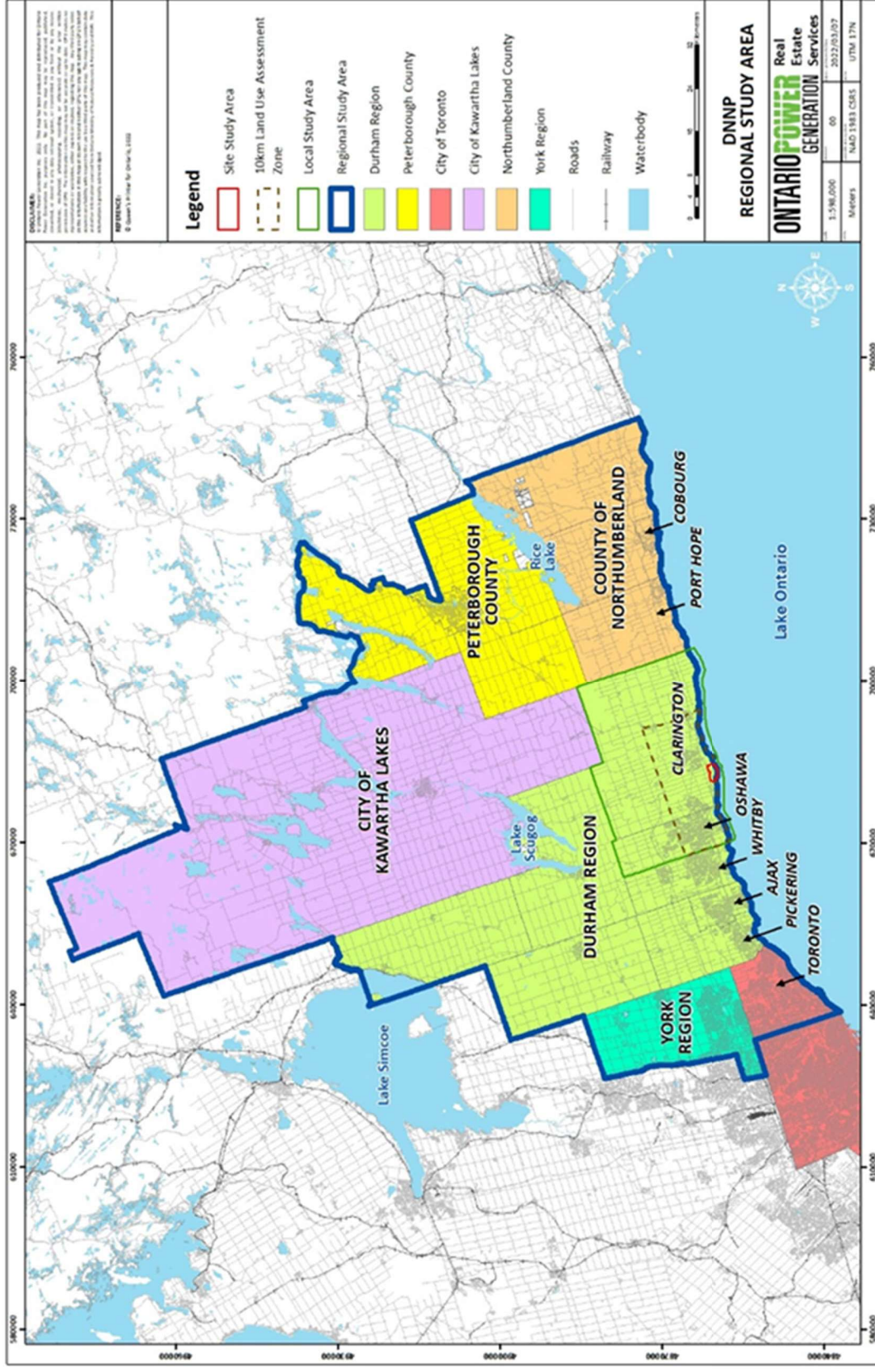


Figure 2.1.1-4: DNNP Regional Study Area



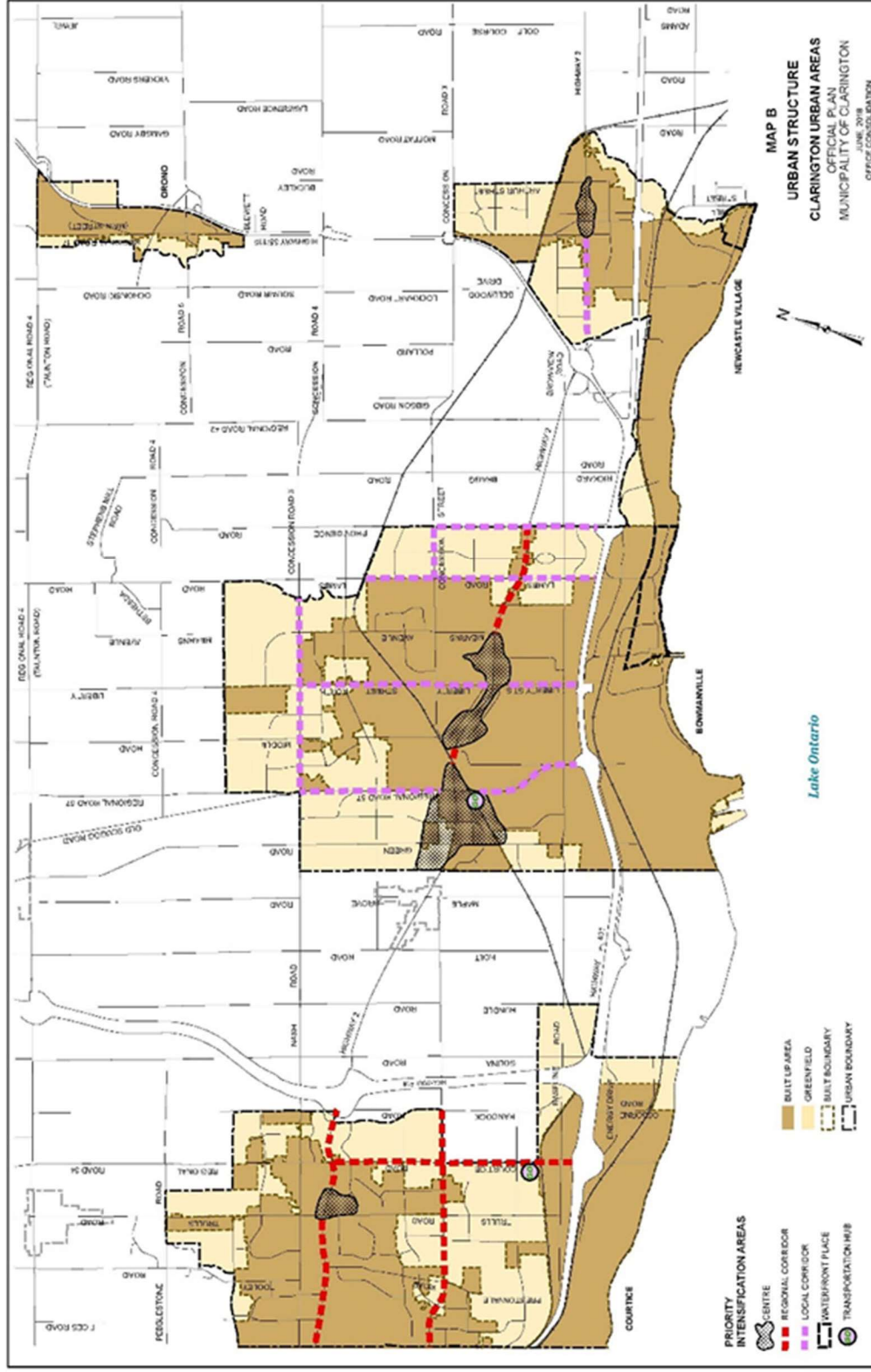


Figure 2.1.3-1: Clarington Urban Areas



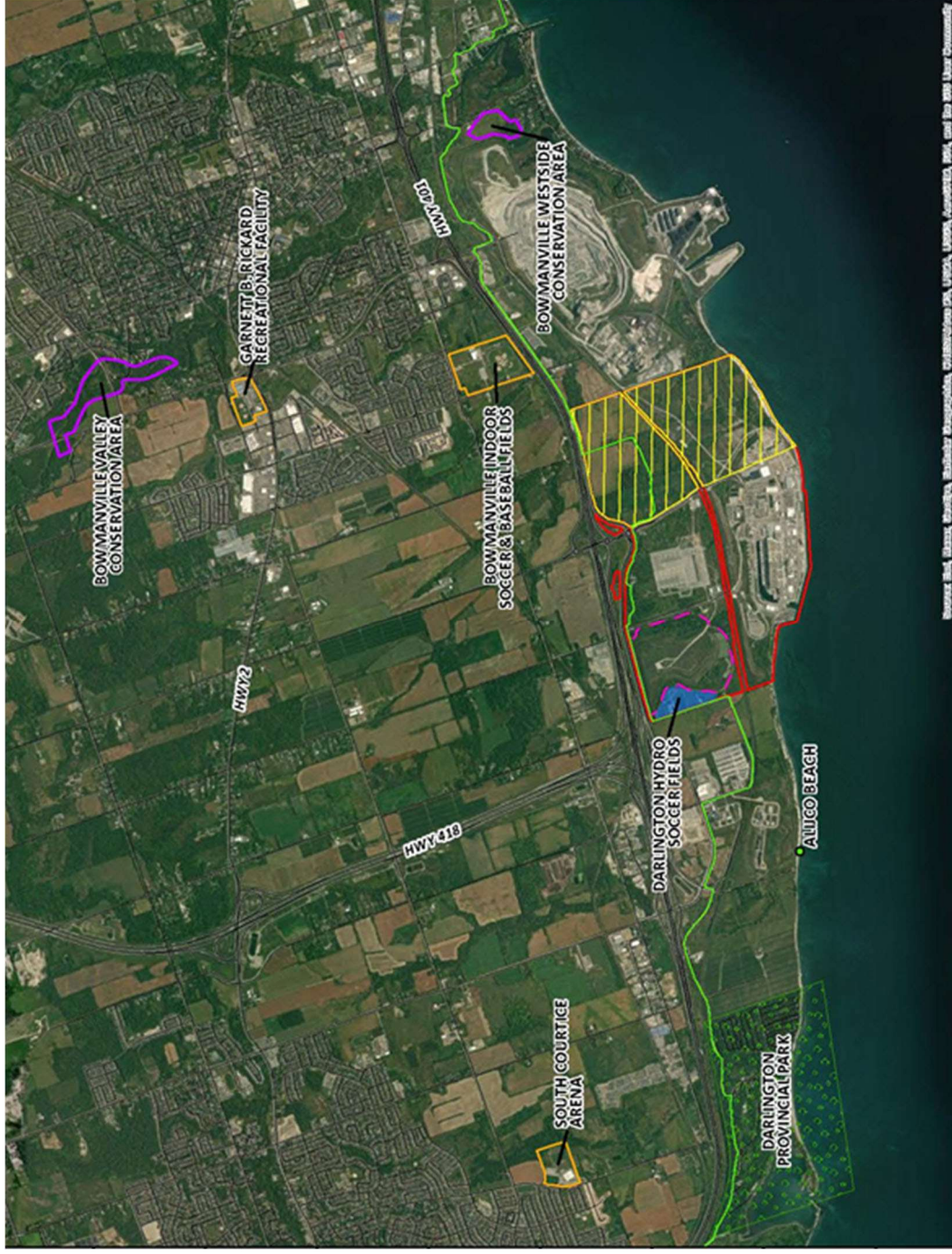


Figure 2.1.8-1: Darlington Nuclear Site – Active Darlington Waterfront Trail

## **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.

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**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.

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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.
2.2.4 Characterization of Stationary Non-Nuclear Accidents			
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.
2.2.5 Characterization of Stationary Nuclear Accidents Hazards			
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.	
2.2.5.4 Characterization of Other Radiological Hazards from DNGS			
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.	

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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.

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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  $< 1.0\text{E-}6/\text{yr}$ . Applies only to reactor sources and not to non-reactor sources.
- QN-2: Design Basis Hazard Frequency,  $< 1.0\text{E-}5/\text{yr}$  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)  $< 1.0\text{E-}7/\text{yr}$  is considered.
- QN-4: Design Basis Hazard Frequency,  $< 1.0\text{E-}6/\text{yr}$  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,  $< 1.0\text{E-}6/\text{yr}$  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  $< 1.0\text{E-}7/\text{yr}$ . 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



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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).

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Similarly, the explosion hazard from a Canadian National Railway derailment accident near the DNNP site has an estimated frequency of  $9.0E-07$  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.

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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

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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.

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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).



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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.

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**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.

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

Chemical/Material (Formula/Trade/State)	Location (subject to change)	Quantity	Hazard Screening
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.

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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.

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**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		

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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.

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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 9A1.3, 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

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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.

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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.



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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

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- 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).

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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)

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- 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

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- 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.

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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.

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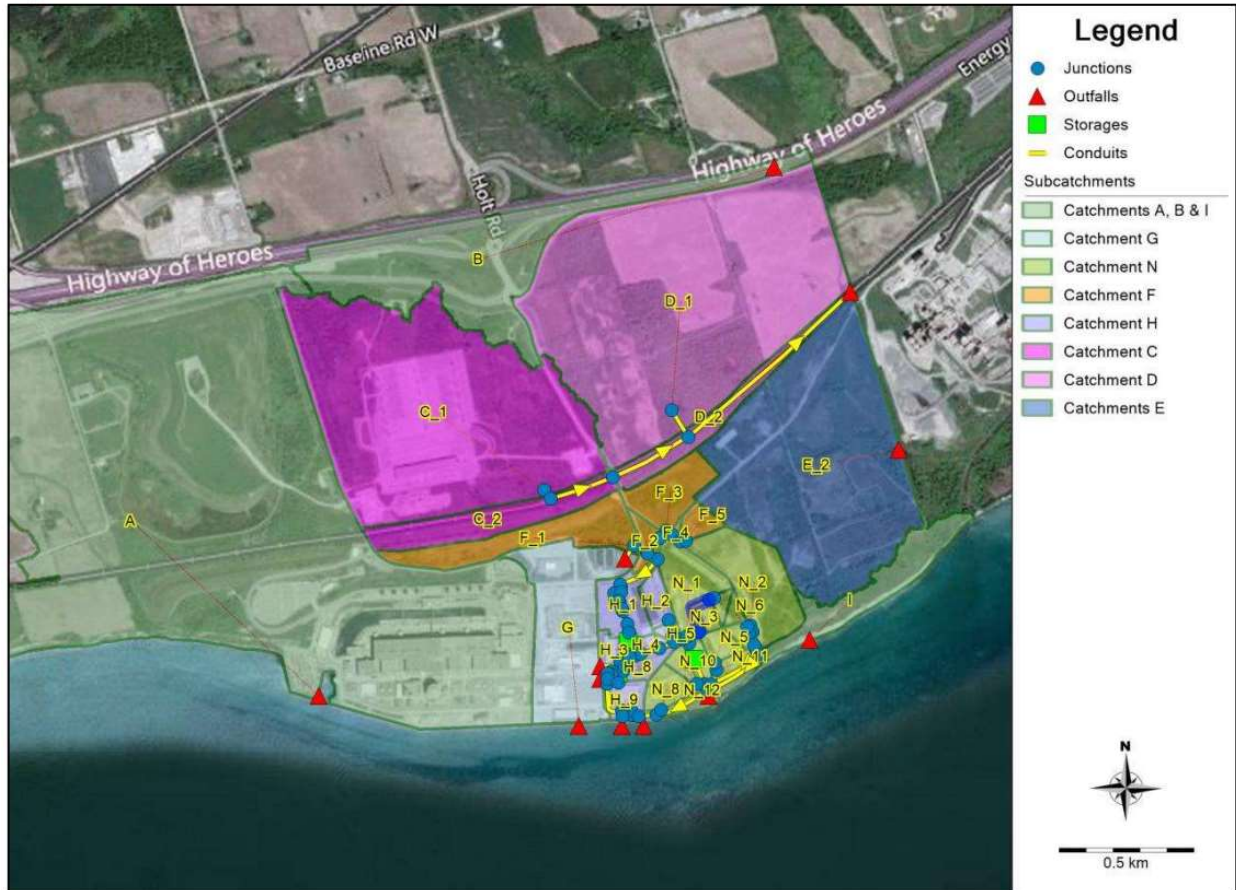


Figure 2.5.3.2-1 Pre-development Darlington Nuclear Site Drainage (Reference 2.5-18)



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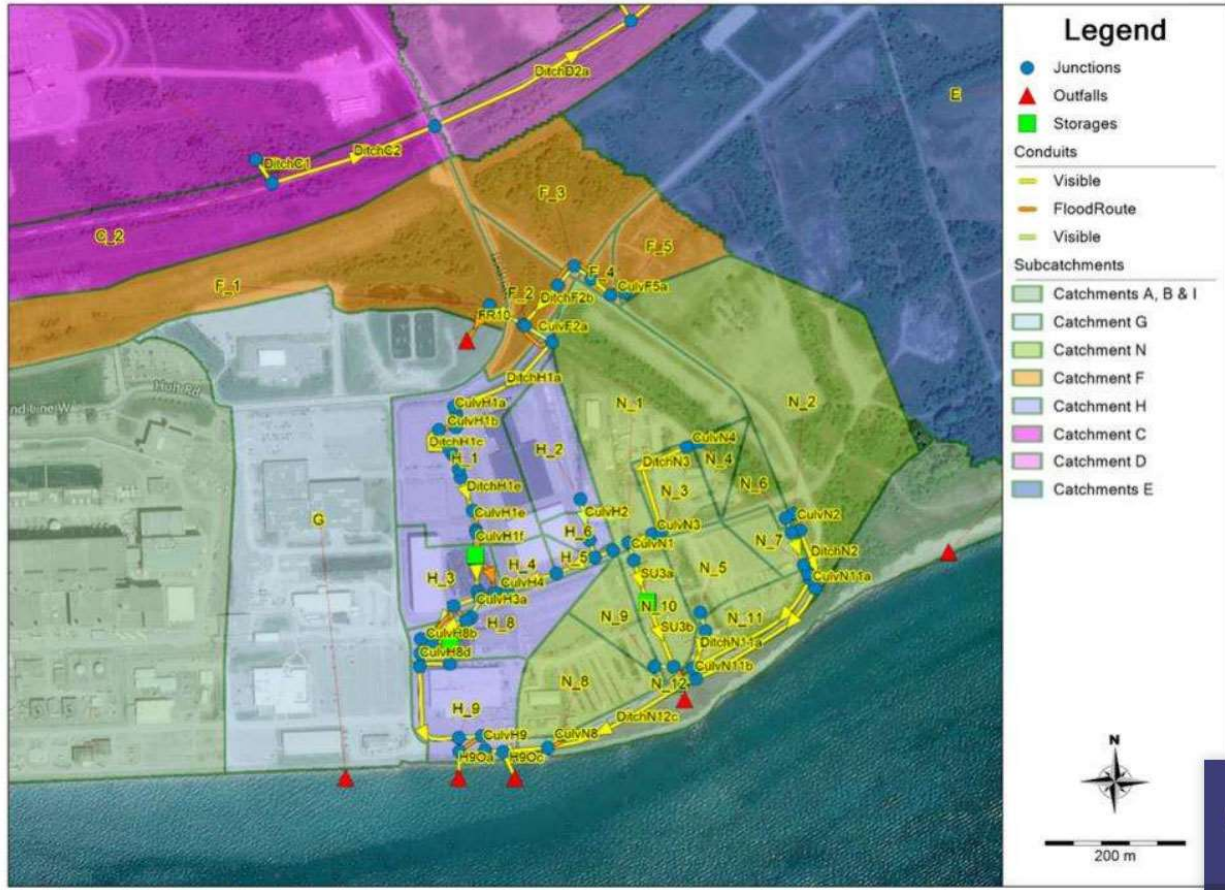


Figure 2.5.3.2-2 Post-development Darlington Nuclear Site Drainage (Reference 2.5-18)



### 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.

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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.

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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).

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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

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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>.
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- 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.



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- 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.

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**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).				

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**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

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**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,

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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

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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.



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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%.

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**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

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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

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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.

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**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)

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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).

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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.

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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.

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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



**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 DNPP 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

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- 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

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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.



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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

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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.

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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

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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.



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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

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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 ( $q_u$ ):

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- 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 (qa) 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 (Ksp) 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

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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.

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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-01210-00175 (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.

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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.



**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.	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

**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

**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.

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- 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.
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**Legend**

- OPG Ownership
- Hydro One Switchyard
- Building Footprint
- Contour Line - 2m Interval\*
- Protected Area Fence
- Utility Power Lines
- Tower
- Waterbody
- Infrastructure
- Road
- Railway

**OPG NEW NUCLEAR AT DARLINGTON EXISTING TOPOGRAPHIC DRAWING FOR ILLUSTRATION PURPOSES**

**ONTARIO POWER GENERATION** Real Estate Services

**2019/12/12** D. COYE NAD 1983 CSRS  
1:10,000 NEWNUCDEV  
Meters N. BRYAN NAD 1983 CSRS / COGNET

**LAKE ONTARIO**

**CRAGO ROAD**  
**PARK ROAD**  
**HYDRO ONE BOWMANVILLE SWITCHYARD**  
**DWME**  
**DWMS 4-UNIT STATION**

**485610.00**  
**486040.00**  
**485580.00**

**661400**  
**662000**  
**662600**  
**663200**  
**663800**  
**664400**

2-127



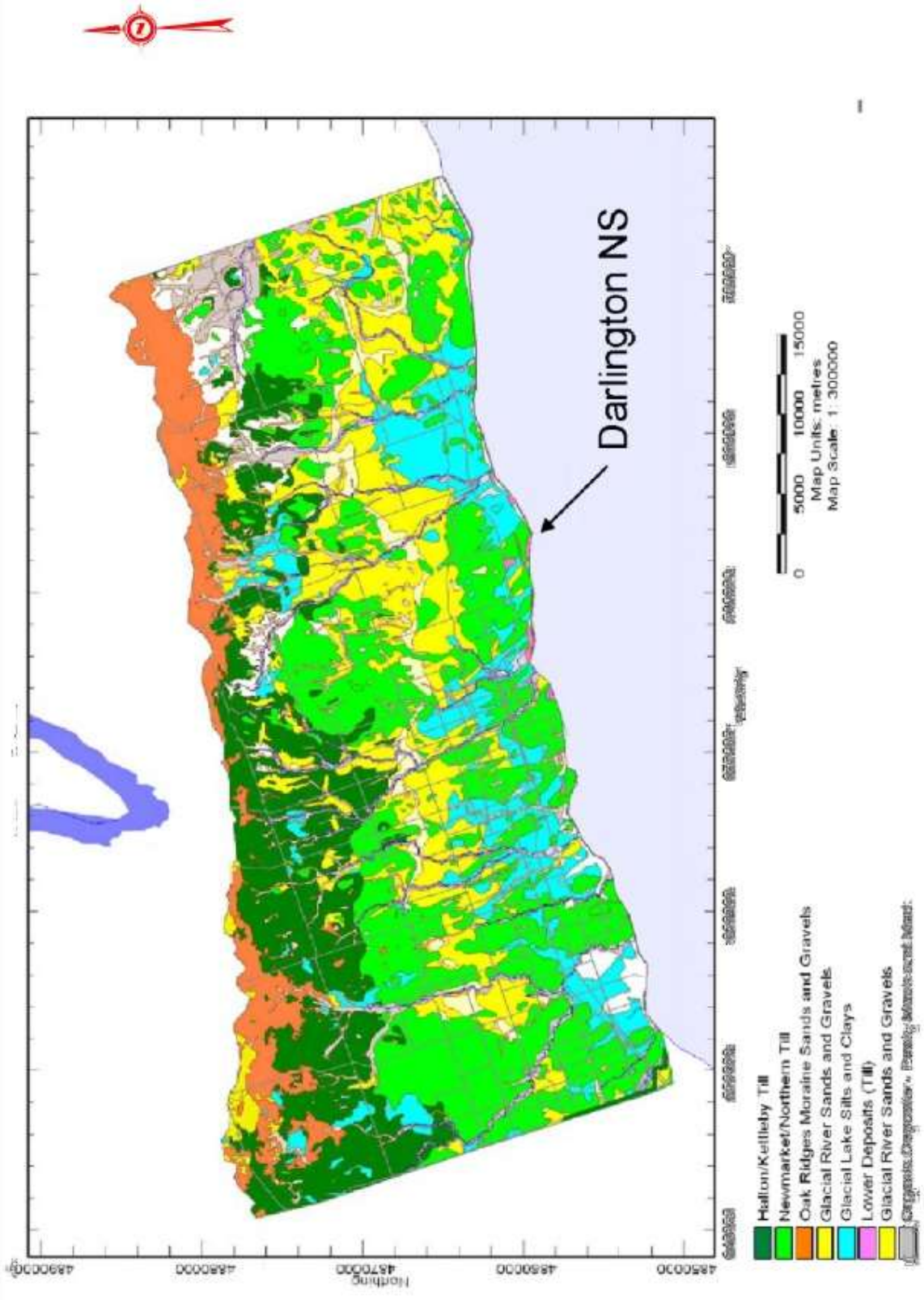


Figure 2.7.2-1: Darlington Nuclear Site - Regional Surficial Geology (Reference 2.7-1)

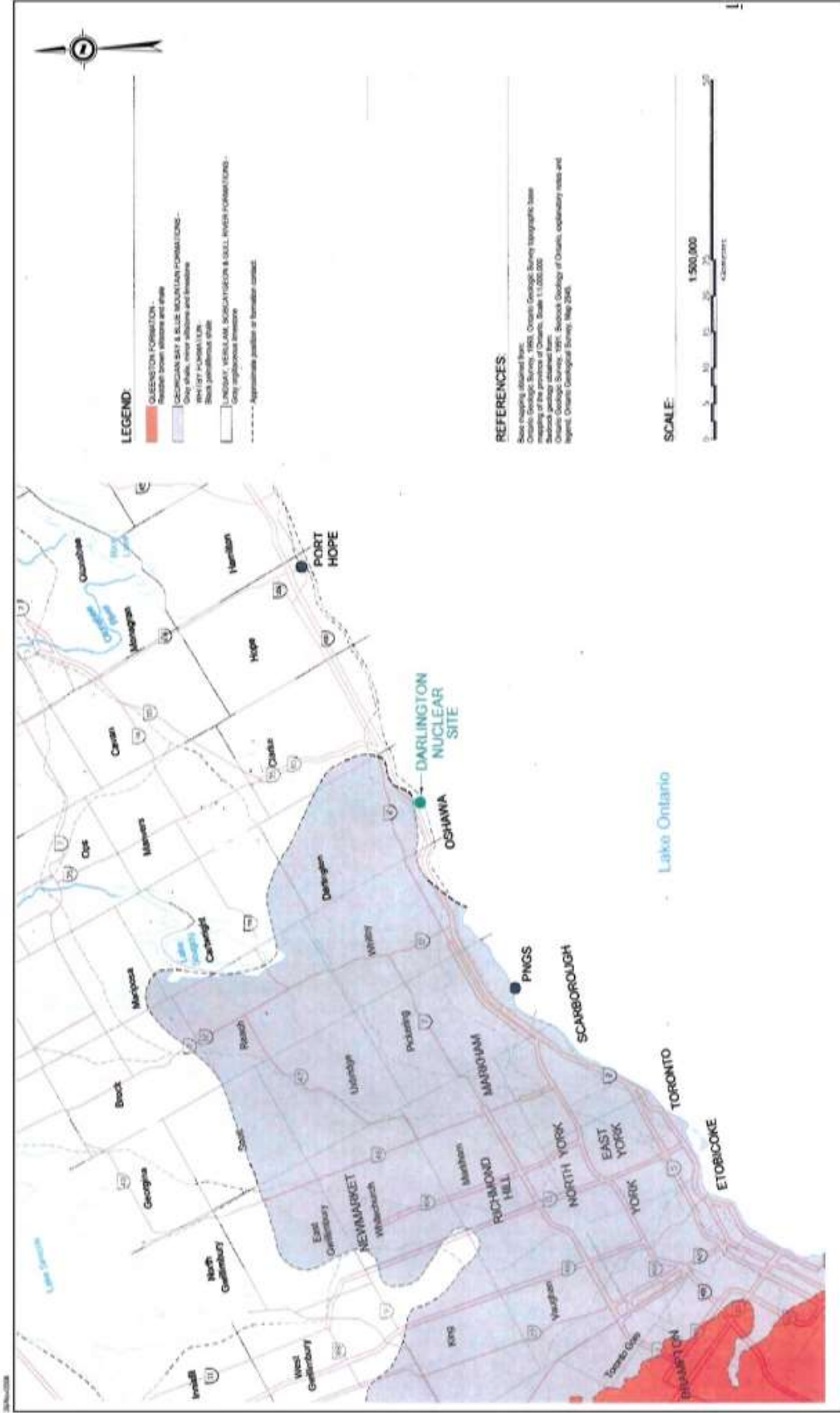


Figure 2.7.2-2: Darlington Nuclear Site - Regional Bedrock Geology (Reference 2.7-1)

[illegible]

2-130



The figure is a topographic map of the Bowmansville area, showing a yellow triangular study area. The map includes contour lines, a north arrow, and a scale bar. The axes are labeled with Northing (m) and Easting (m) coordinates.

**Legend:**

- PROJECT LOCATION
- LAKE ONTARIO

**NOTES:**

1. GEOPHYSICAL INTERPRETATIONS MAY BE INACCURATE WHERE TARGETS OF INTEREST ARE TOO SMALL OR HAVE SHALLOW DEPTHS.
2. THE DATA WERE ACQUIRED USING A GEOPHYSICAL INSTRUMENT THAT IS CALIBRATED TO THE CANADIAN SYSTEM OF UNITS (SI) AND MAY NOT BE ACCURATE FOR OTHER SYSTEMS.
3. THE DATA WERE ACQUIRED USING A GEOPHYSICAL INSTRUMENT THAT IS CALIBRATED TO THE CANADIAN SYSTEM OF UNITS (SI) AND MAY NOT BE ACCURATE FOR OTHER SYSTEMS.
4. THE DATA WERE ACQUIRED USING A GEOPHYSICAL INSTRUMENT THAT IS CALIBRATED TO THE CANADIAN SYSTEM OF UNITS (SI) AND MAY NOT BE ACCURATE FOR OTHER SYSTEMS.
5. THE DATA WERE ACQUIRED USING A GEOPHYSICAL INSTRUMENT THAT IS CALIBRATED TO THE CANADIAN SYSTEM OF UNITS (SI) AND MAY NOT BE ACCURATE FOR OTHER SYSTEMS.

**REFERENCES:**

1. PROJECTOR TRANSVERSE MERCATOR DATUM AND MECS COORDINATE SYSTEM DATUM 27N

**CLIENT:**  
E.S. FOX LIMITED NUCLEAR SERVICES

**PROJECT:**  
OFFSHORE GEOPHYSICAL INVESTIGATION: DARTINGTON NEW NUCLEAR PROJECT, BOWMANVILLE, ONTARIO

**TITLE:**  
OFFSHORE GEOPHYSICS BATHYMETRIC CONTOUR MAP

**CONSULTANT:**  
TYT-Y-AM-00 2023-01-16  
PREPARED: FAW  
DESIGN: FAW  
REVIEW: CP  
APPROVED: CP

**PROJECT NO.:** 21451329

**PHASE:** 0

**DATE:** 2023-01-16

**SCALE:** 1:5000

**UNIT:** METERS

**COORDINATE SYSTEM:** UTM 27N

**PROJECTION:** TRANSVERSE MERCATOR

**DATUM:** CANADIAN DATUM 27N

**MAP SCALE:** 1:5000

**MAP DATE:** 2023-01-16

**MAP BY:** FAW

**MAP CHECKED BY:** CP

**MAP APPROVED BY:** CP

**MAP SCALE BAR:** 0 10 20 30 40 50 60 70 80 90 100 METERS

**MAP NORTH ARROW:** NORTH

**MAP COORDINATES:** NORTHING (m) 4857600 4857800 4858000 4858200 4858400 4858600 4858800 4859000 4859200 4859400 4859600 4859800 4860000  
EASTING (m) 683600 683800 684000 684200 684400 684600 684800 685000 685200 685400 685600 685800 686000

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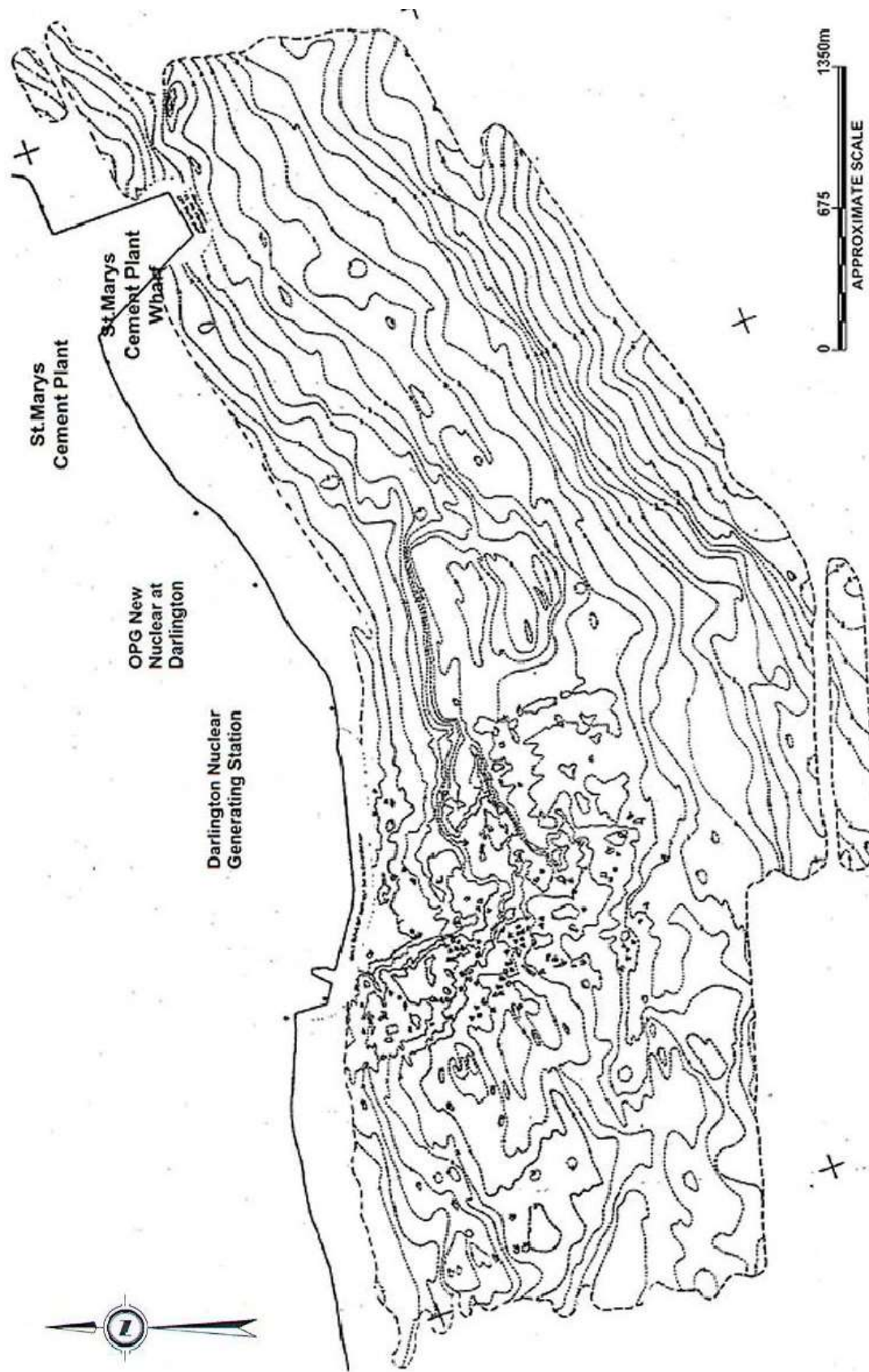


Figure 2.7.2-5: Lakebed Bathymetric Contours along DNNP Site's Shoreline (Reference 2.7-2)



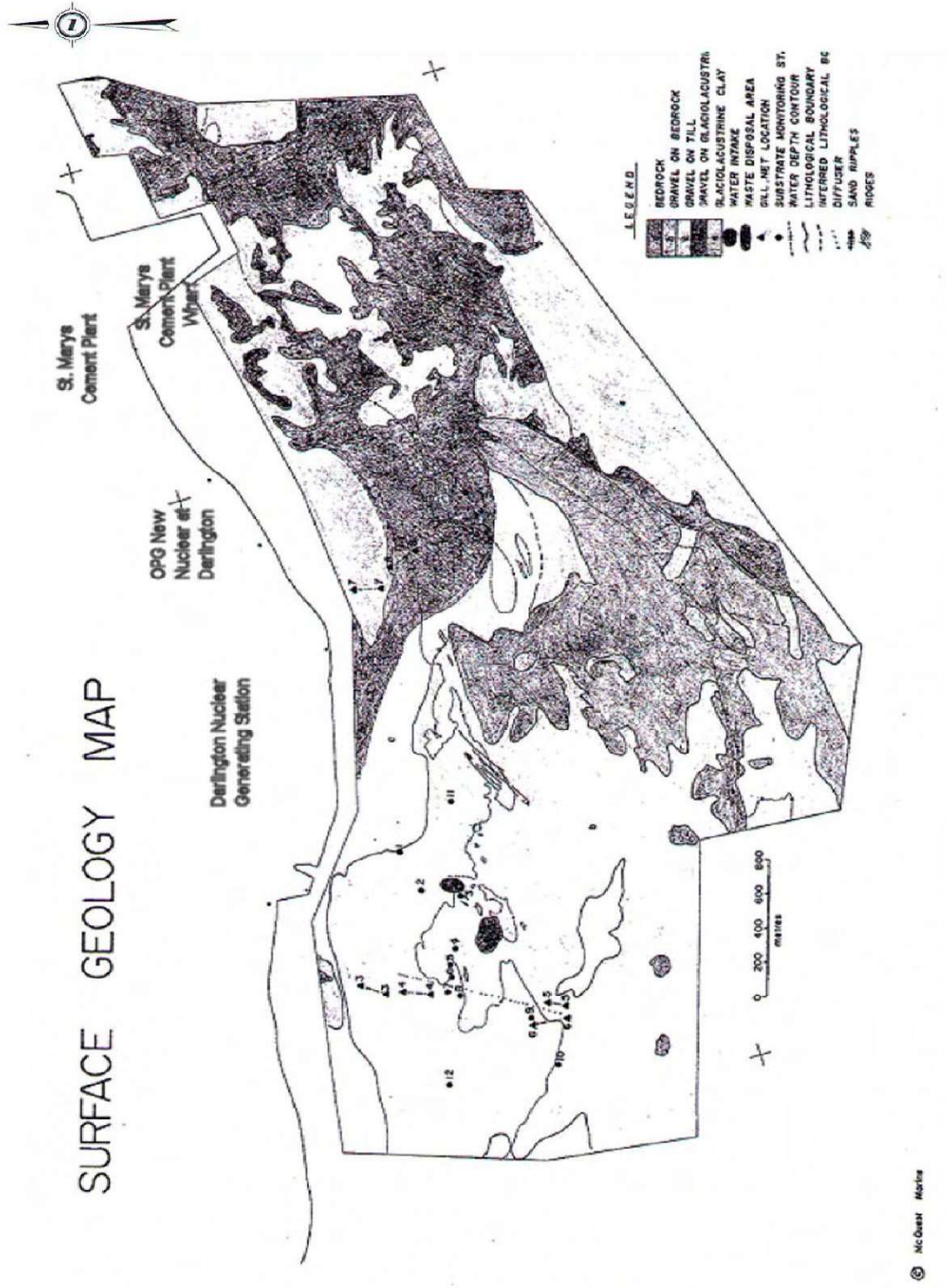


Figure 2.7.2-6: Lakebed Surface Geology Map along DNNP Site's Shoreline (Reference 2.7-2)

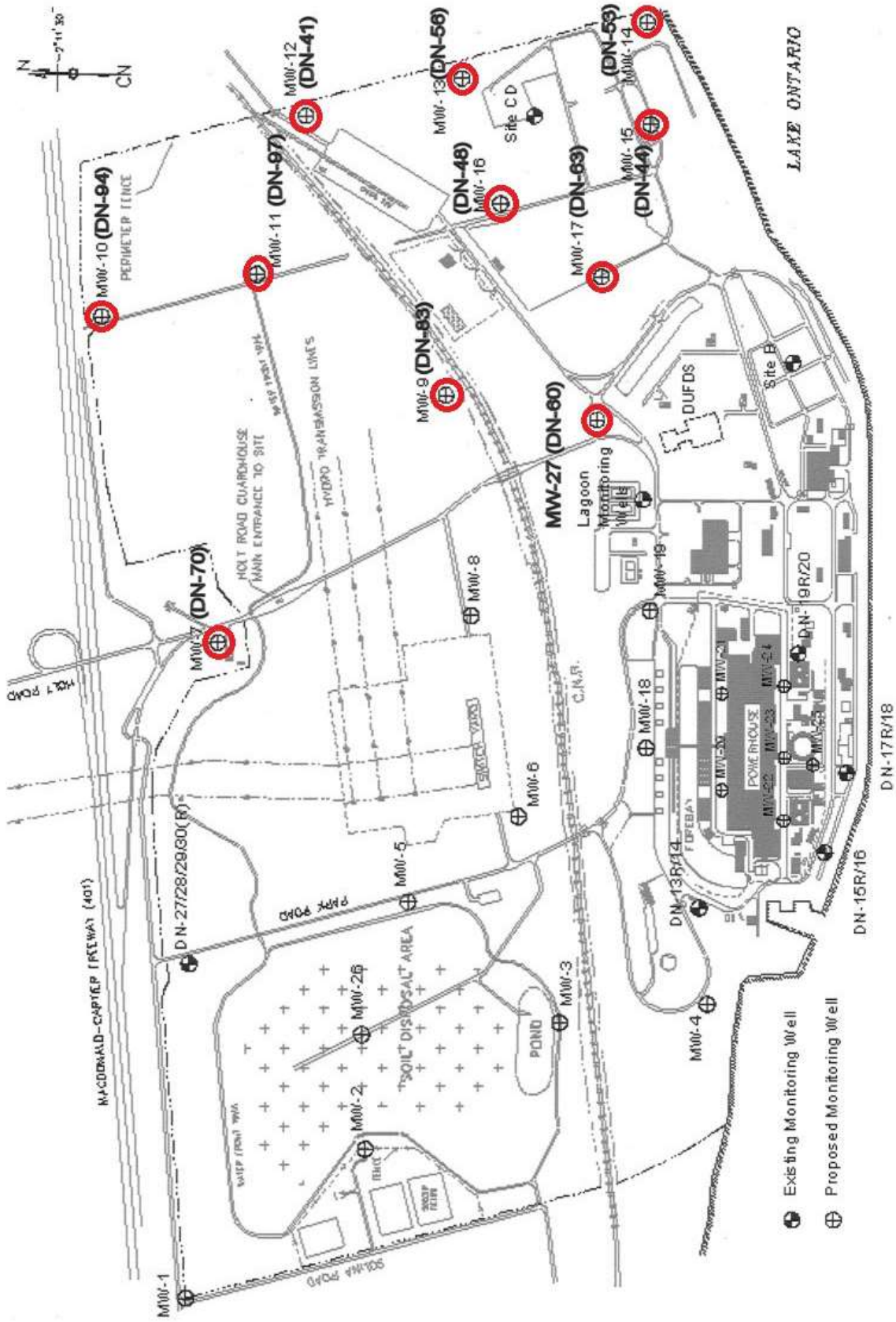


Figure 2.7.3.1-1: Locations of CH2MHILL (2009) Monitoring Wells/Boreholes (Reference 2.7-1)



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**Figure 2.7.3.1-2: Locations of AMEC (2012) Boreholes (Reference 2.7-35)**

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NON-PROPRIETARY INFORMATION

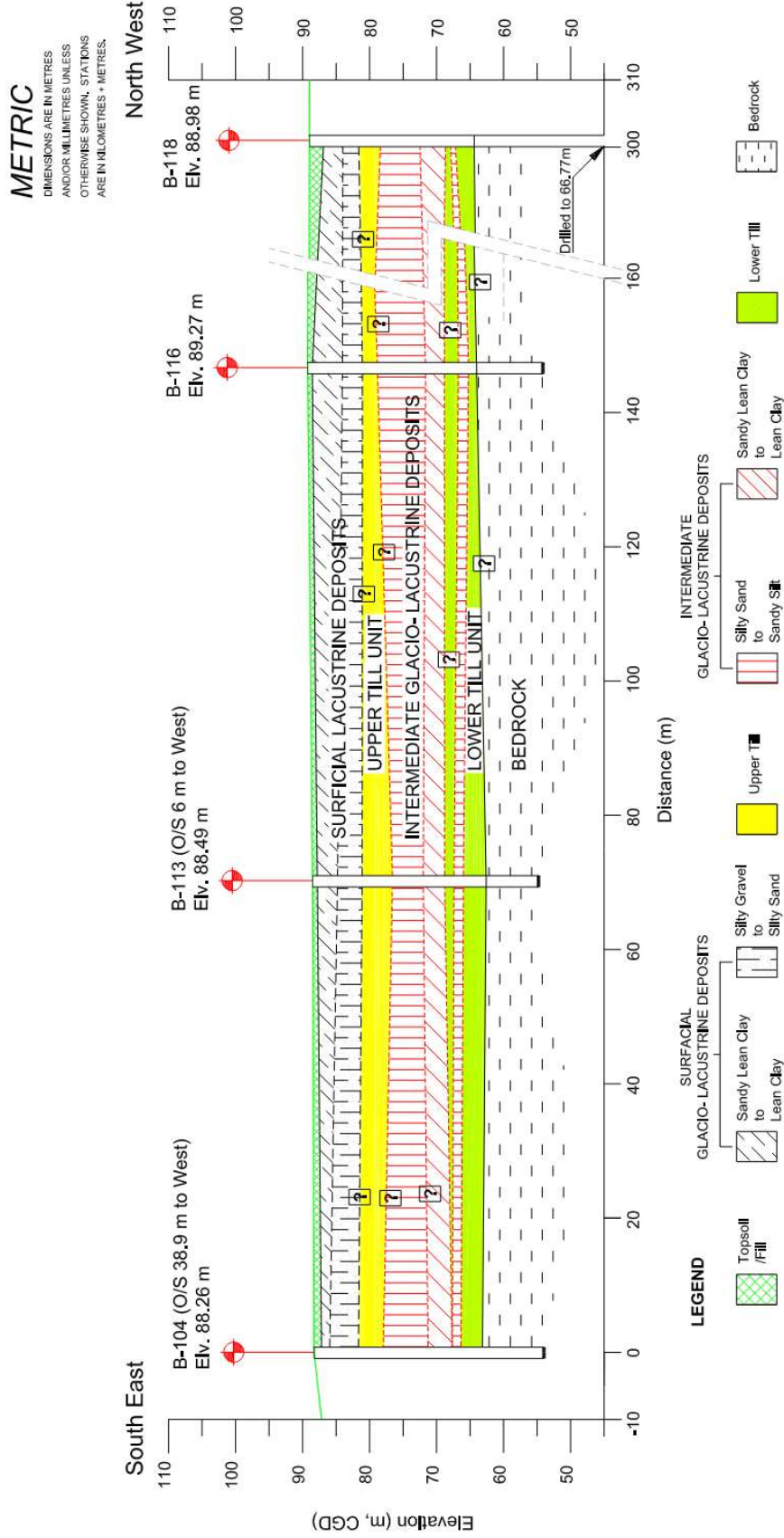


Figure 2.7.3.1-4: Stratigraphic Profile near BWRX-300 Location (Reference 2.7-36)

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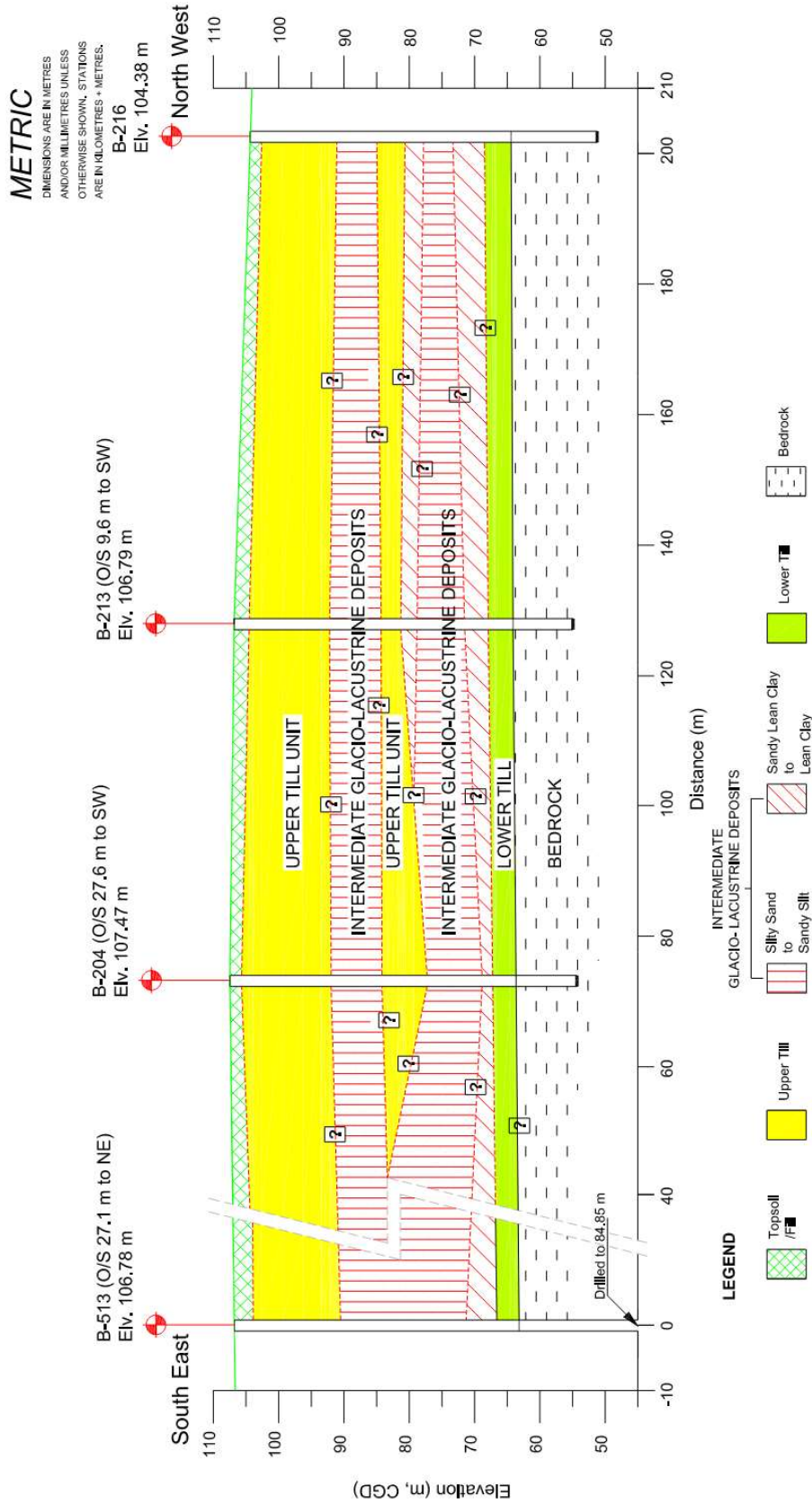


Figure 2.7.3.1-5: Stratigraphic Profile near Topographic Ridge East of the BWRX-300 Location (Reference 2.7-36)



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2-139



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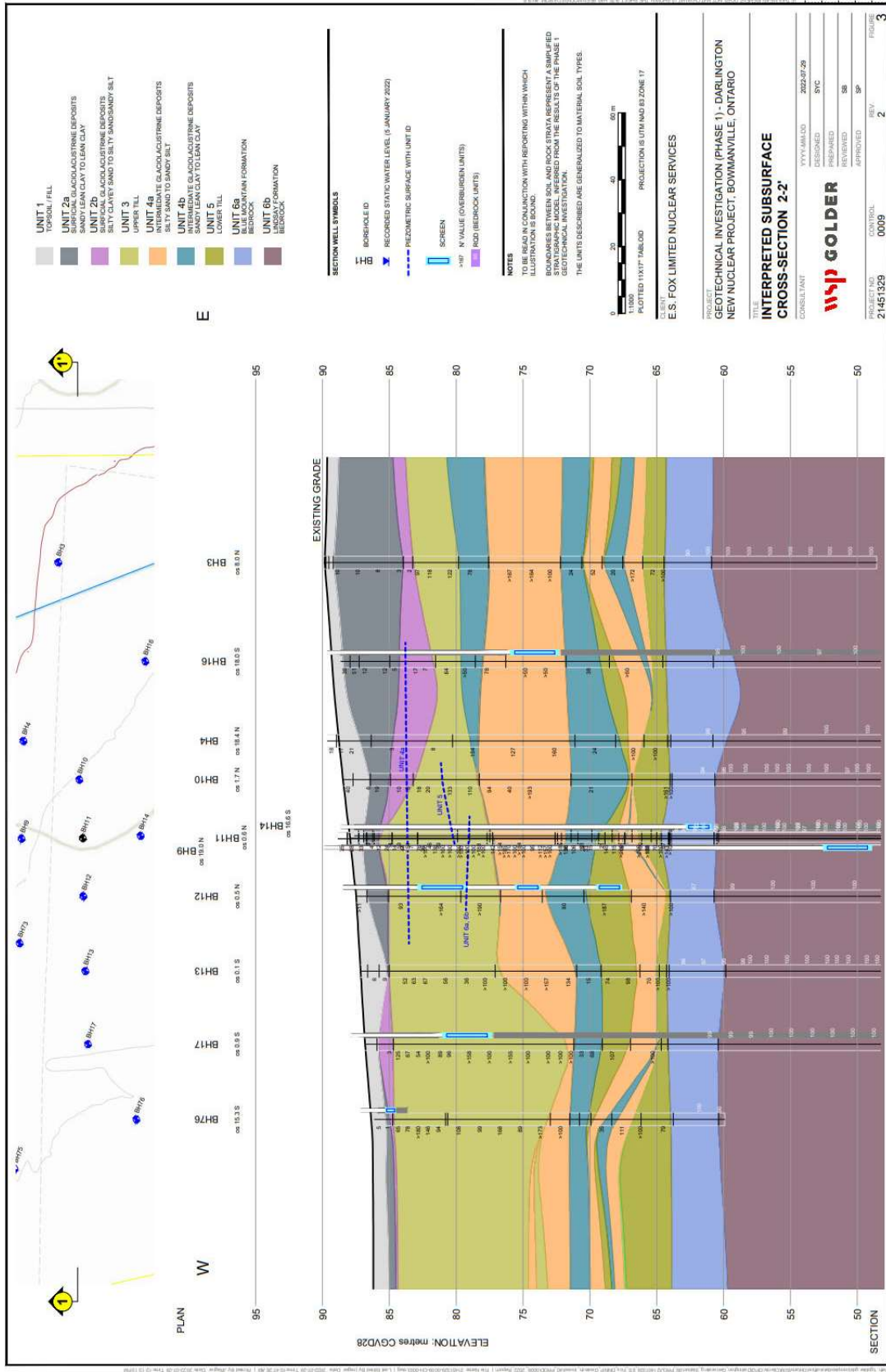


Figure 2.7.3.2-1: Subsurface Stratigraphic Profile at Cross-Section 1-1 (Reference 2.7-39)



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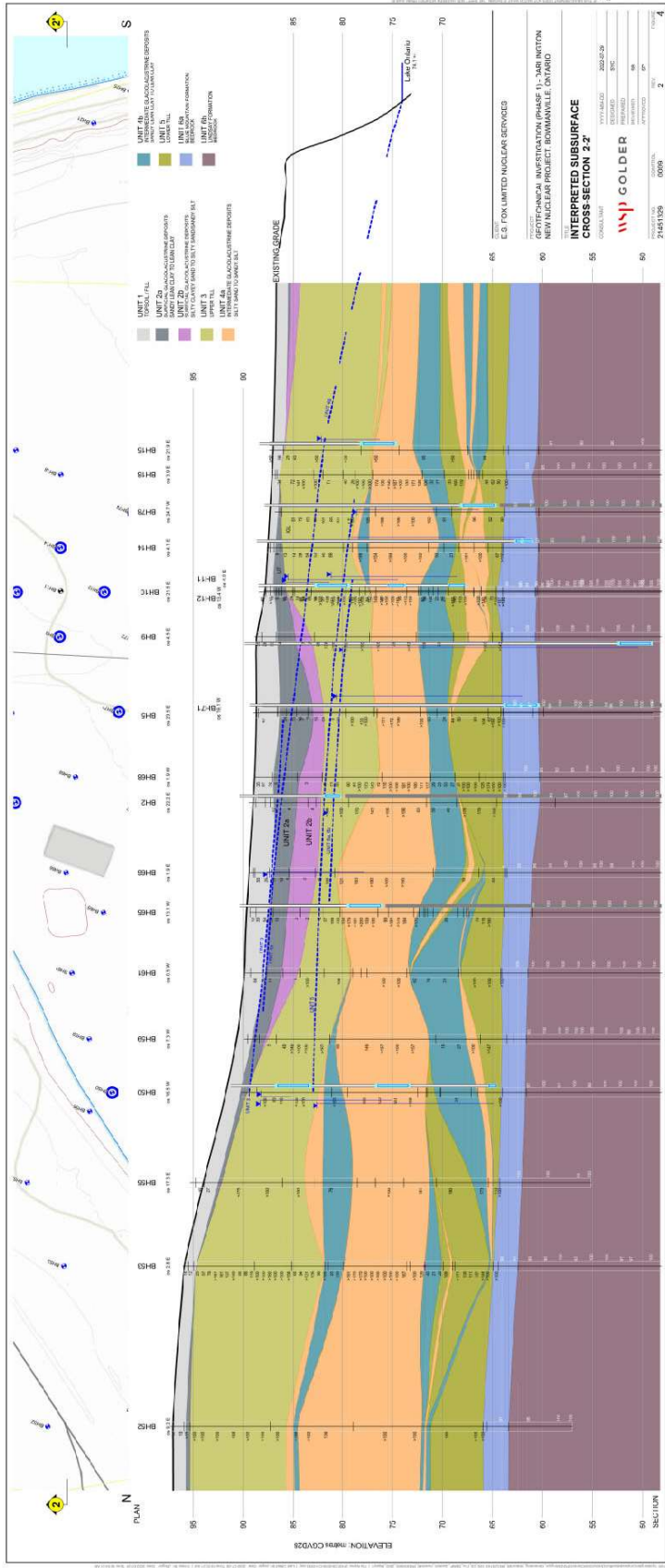


Figure 2.7.3.2-2: Subsurface Stratigraphic Profile at Cross-Section 2-2 (Reference 2.7-39)

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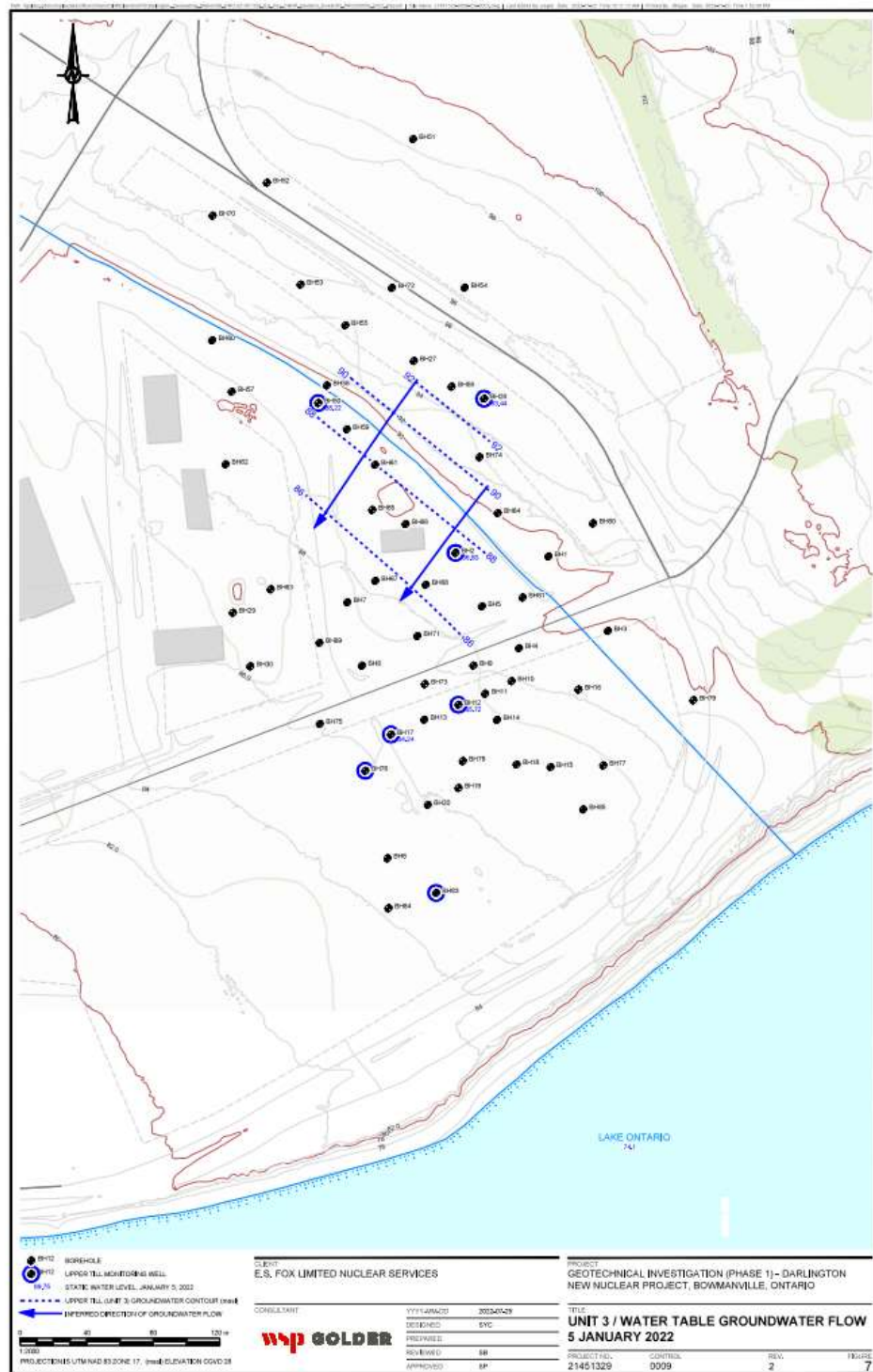


Figure 2.7.3.2-3: Unit 3 Groundwater Levels – Shallow/Water Table (Reference 2.7-39)

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NON-PROPRIETARY INFORMATION

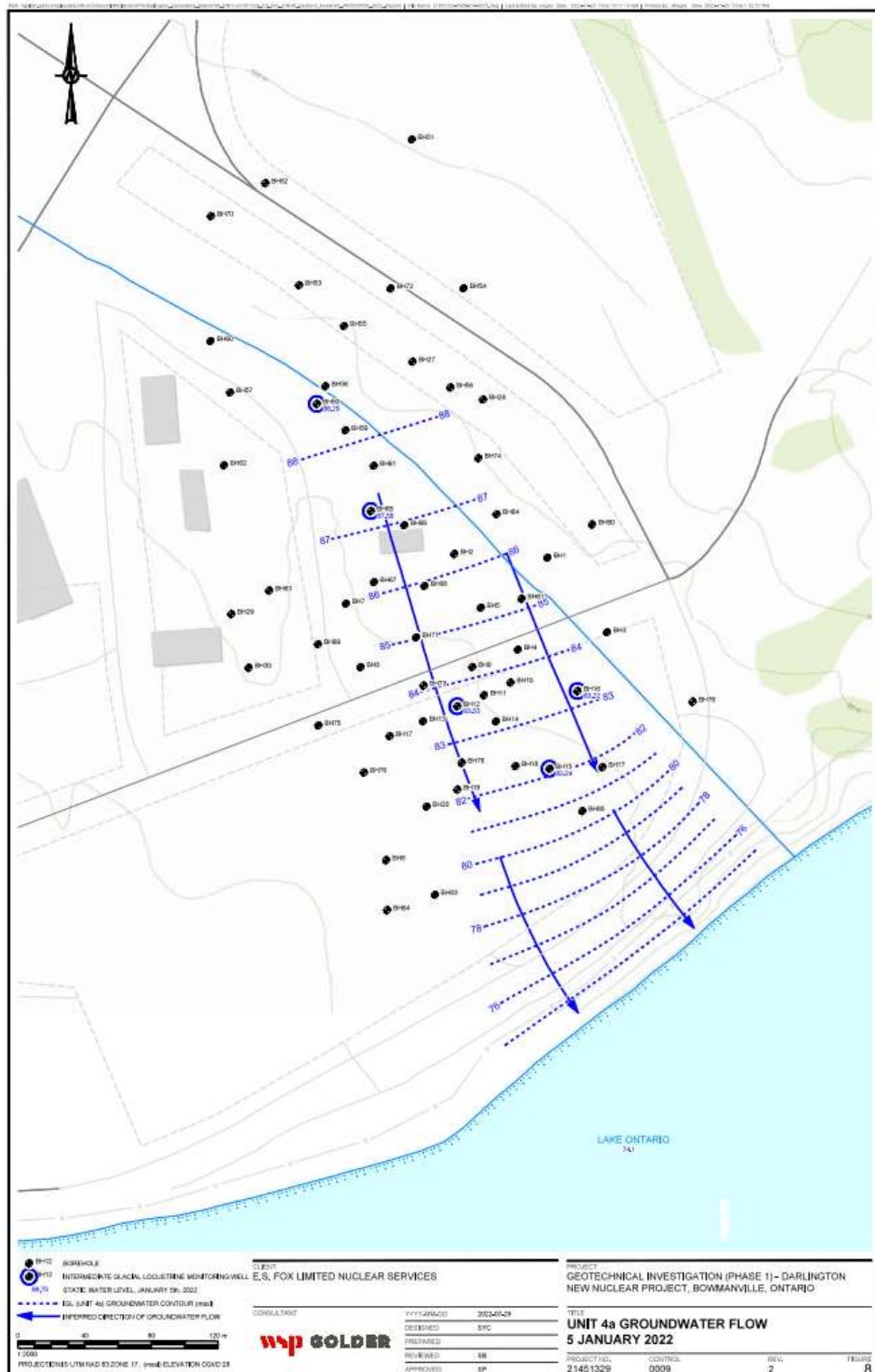


Figure 2.7.3.2-4: Unit 4a Groundwater Flow – Inter-glacial Deposits (Reference 2.7-39)

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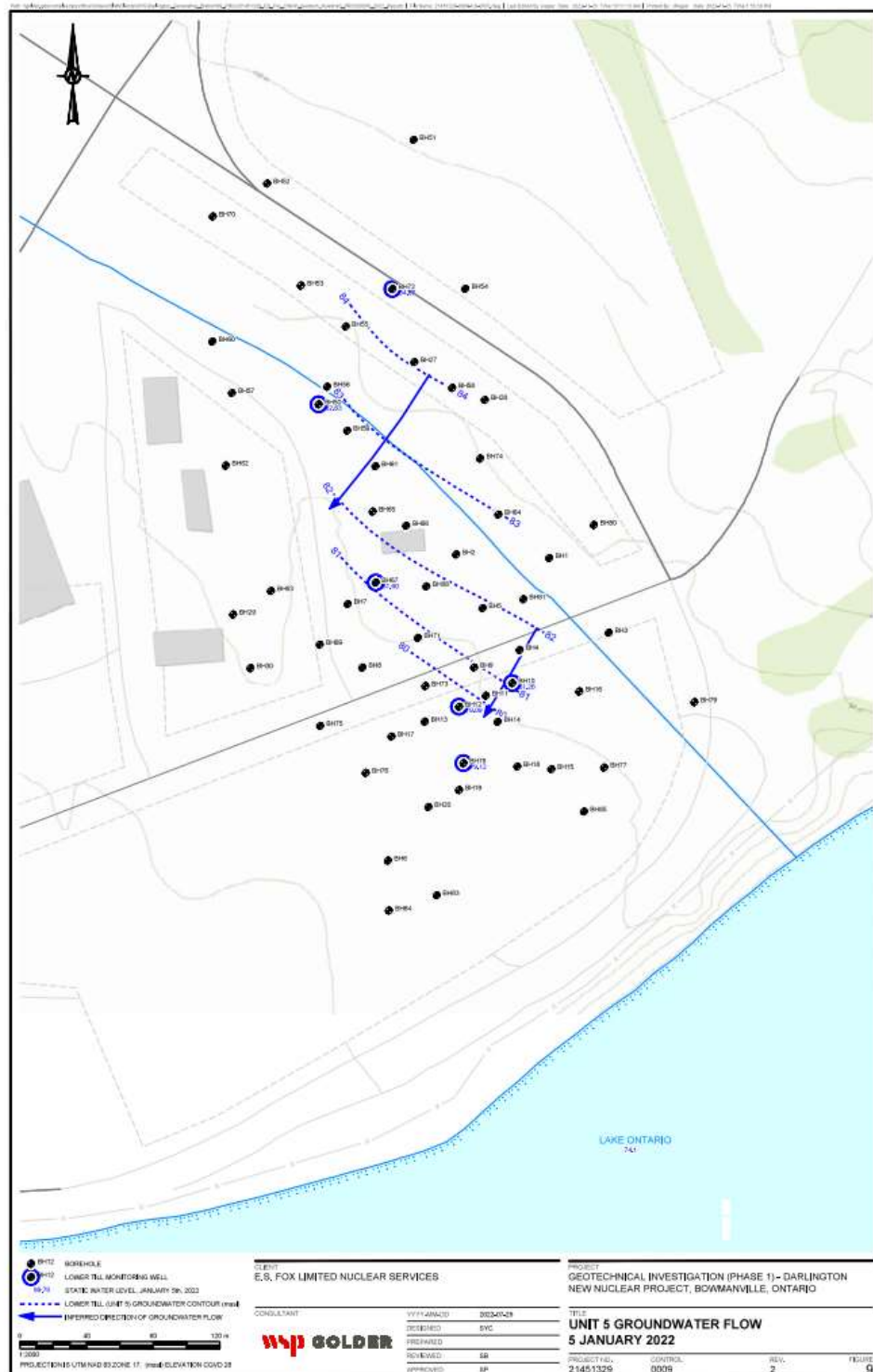


Figure 2.7.3.2-5: Unit 5 Groundwater Flow – Shallow Bedrock (Reference 2.7-39)



2-145

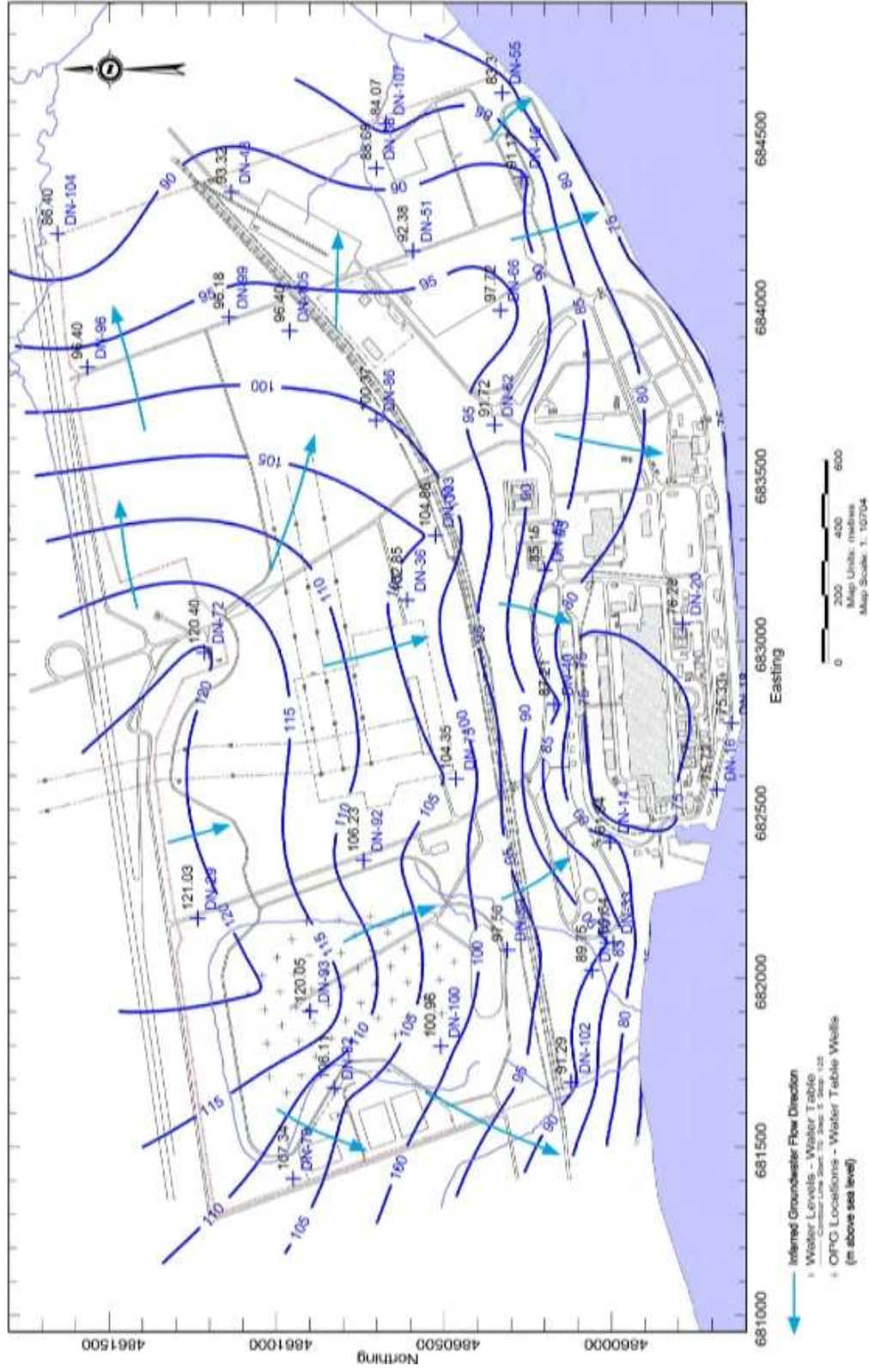


Figure 2.7.3.2-7: Regional Groundwater Levels – Shallow/Water Table (Reference 2.7-1)

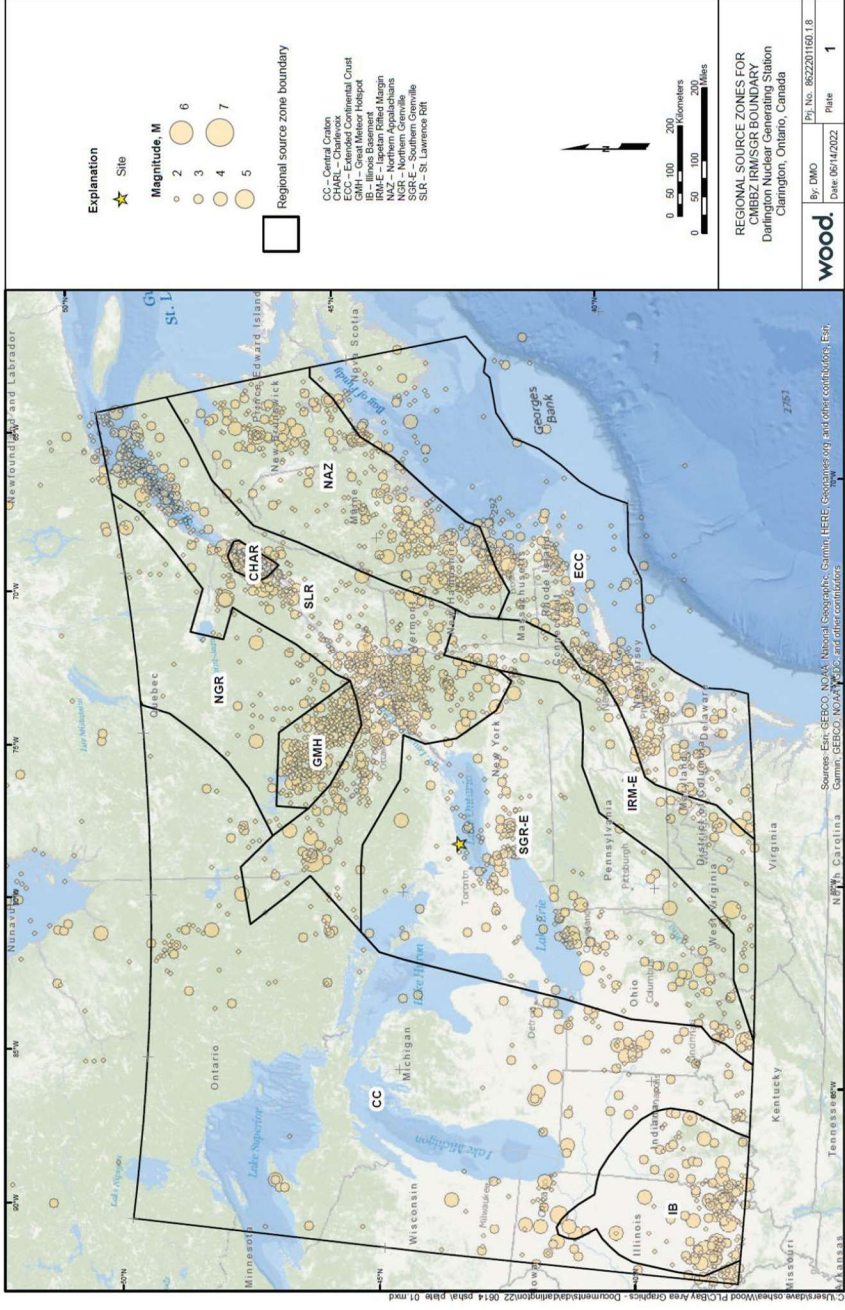
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2-148



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Abbreviations:

CC – Central Craton  
CHAR – Charlevoix  
ECC – Extended Continental Crust  
GMH – Great Meteor Hotspot  
IB – Illinois Basin

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)

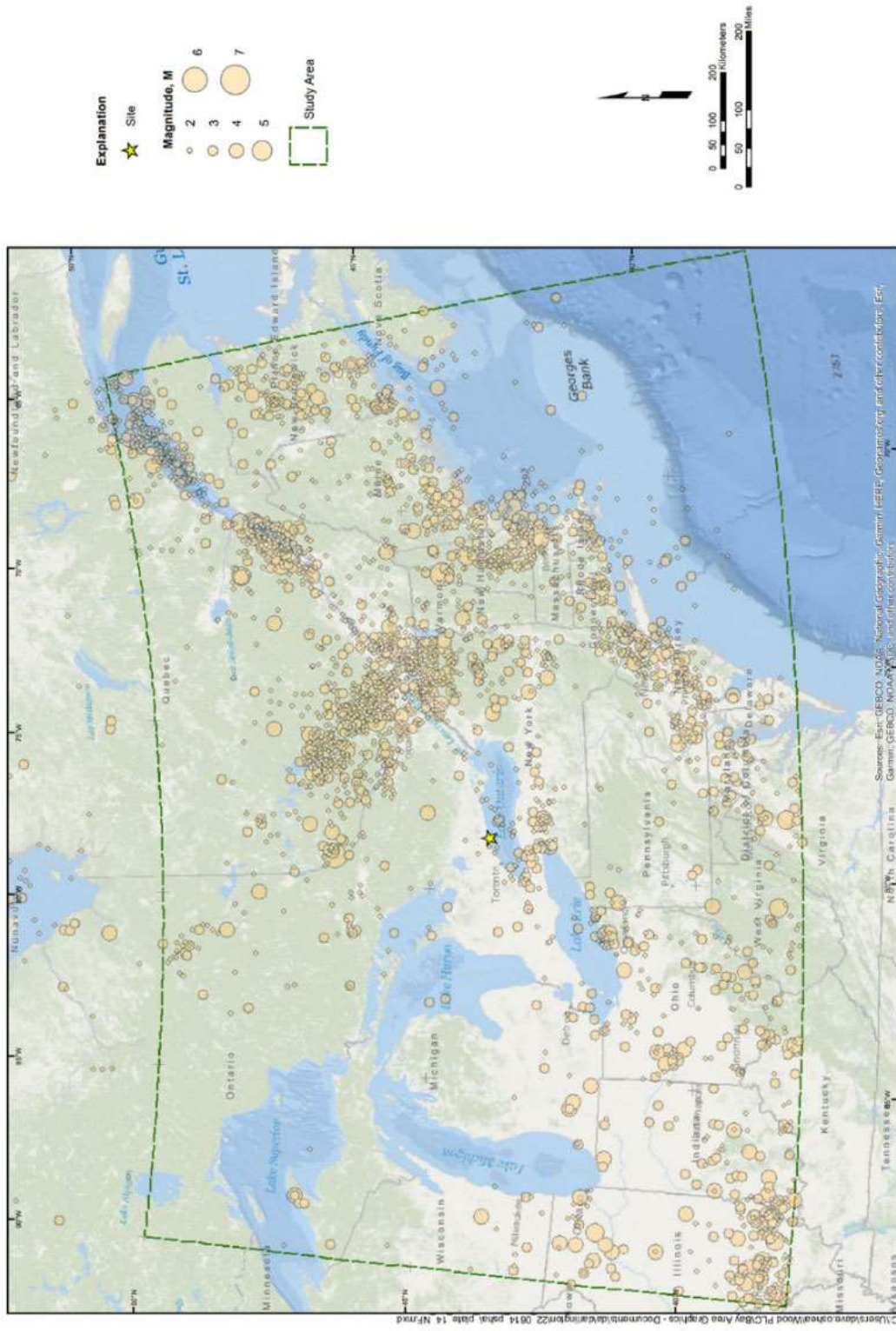


Figure 2.7.4.3-1: Map of Independent Earthquakes in the Updated Earthquake Catalogue for the Study Region (Reference 2.7-41)



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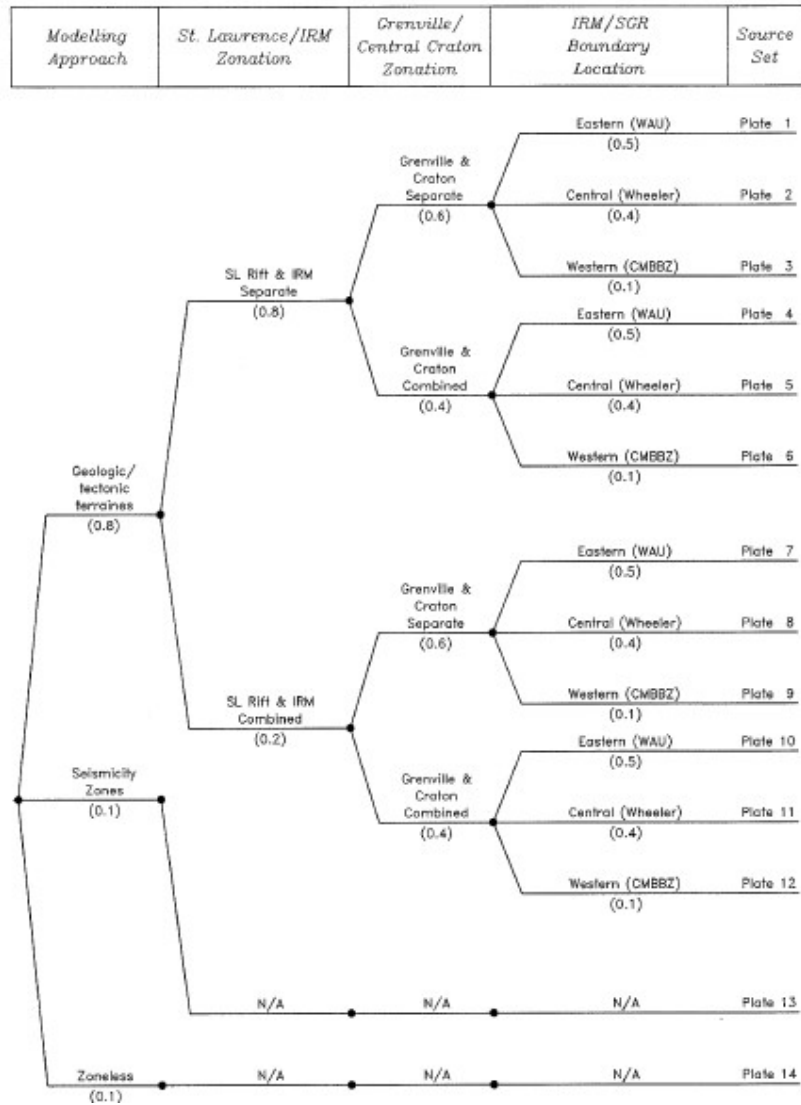


Figure 2.7.4.4-1: Logic Tree for Distributed Seismicity Sources (Reference 2.7- 42)

TIM — Timiskaming  
PMQ — Passamaquoddy Bay  
TRR — Trois-Rivières  
SAG — Saguenay  
SEB — Southeast Canada Background

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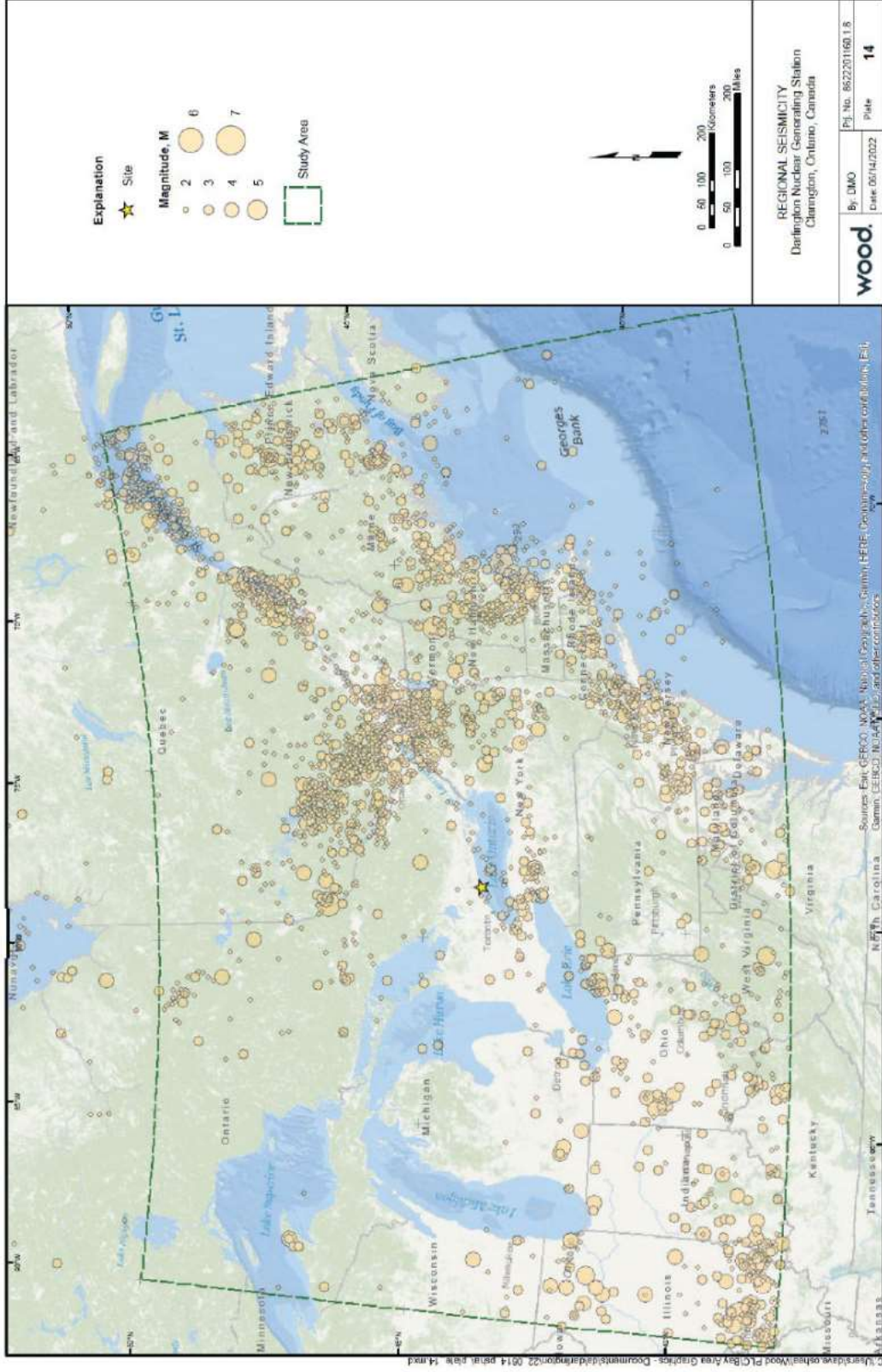
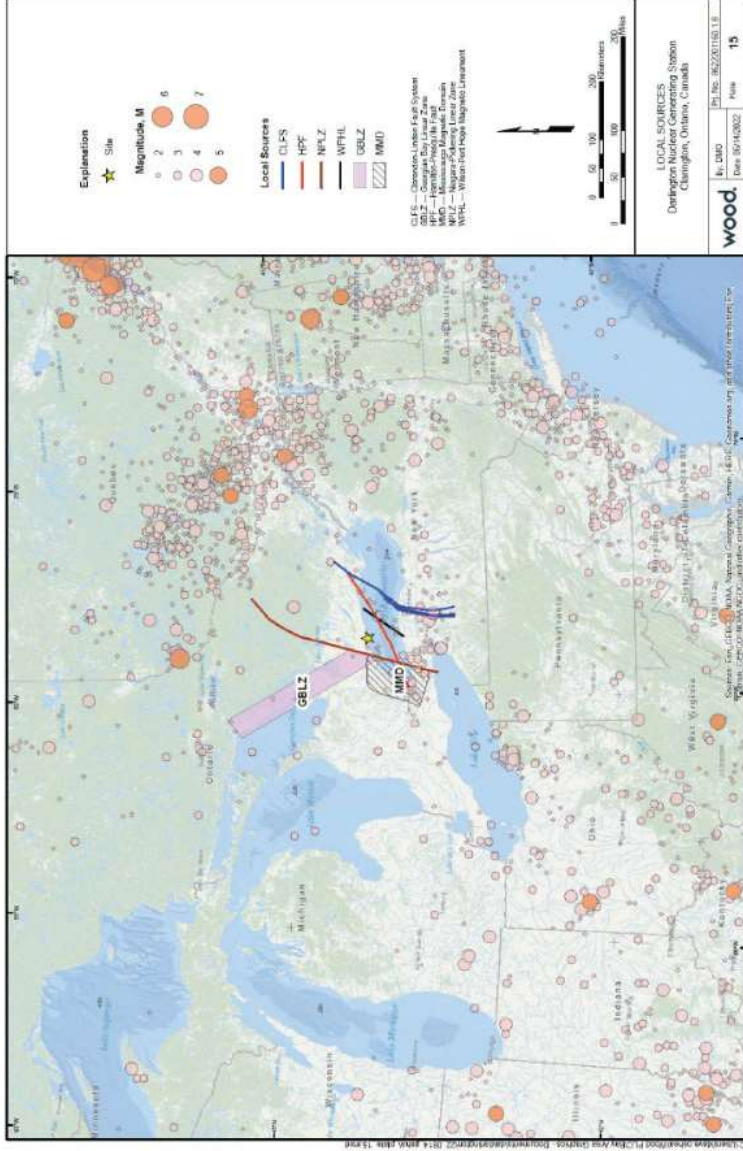


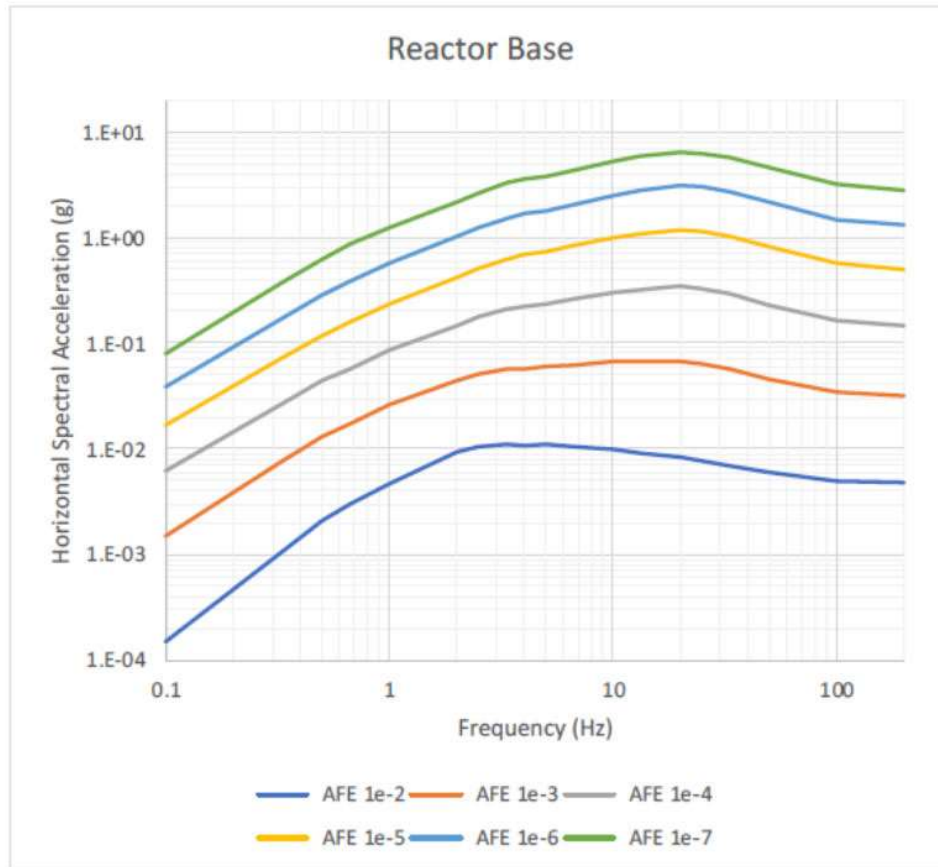
Figure 2.7.4.4-3: Regional Seismicity for Zoneless Model (Plate 14 in Reference 2.7-41)





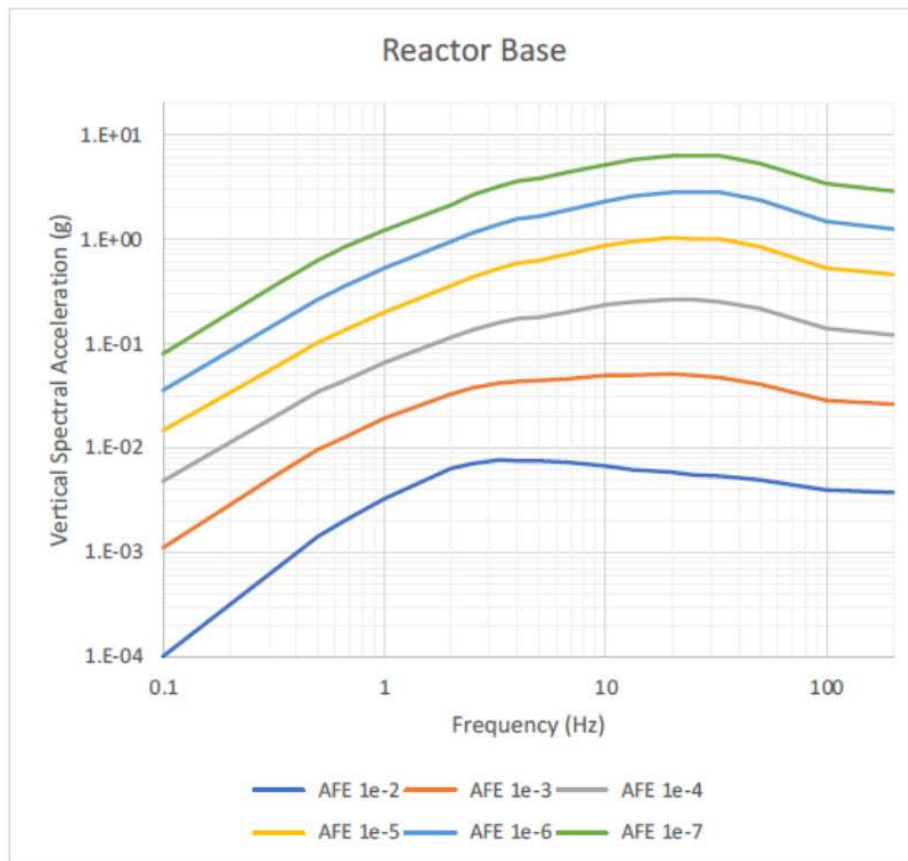
**Abbreviations:**  
CLFS — Clarendon-Linden Fault System  
GBLZ — Georgian Bay Linear Zone  
HPF — Hamilton-Presqu'ile Fault  
MMD — Mississauga Magnetic Domain  
NPLZ — Niagara-Pickering Linear Zone  
WPHL — Wilson-Port Hope Magnetic Lineament

Figure 2.7.4-4: Local Source Zones (Plate 15 in Reference 2.7-41)

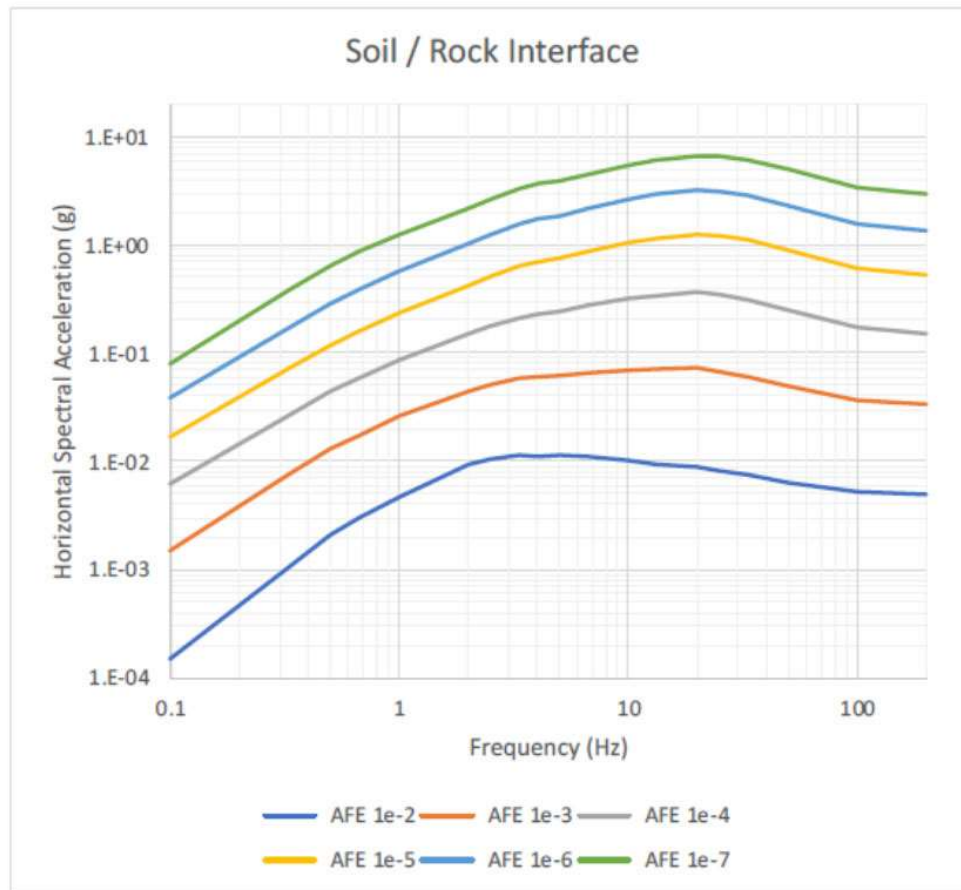


**Figure 2.7.4.6-1: Horizontal UHRS at Elevation 52.93 m Based on Mean Hazard (Reference 2.7-41)**

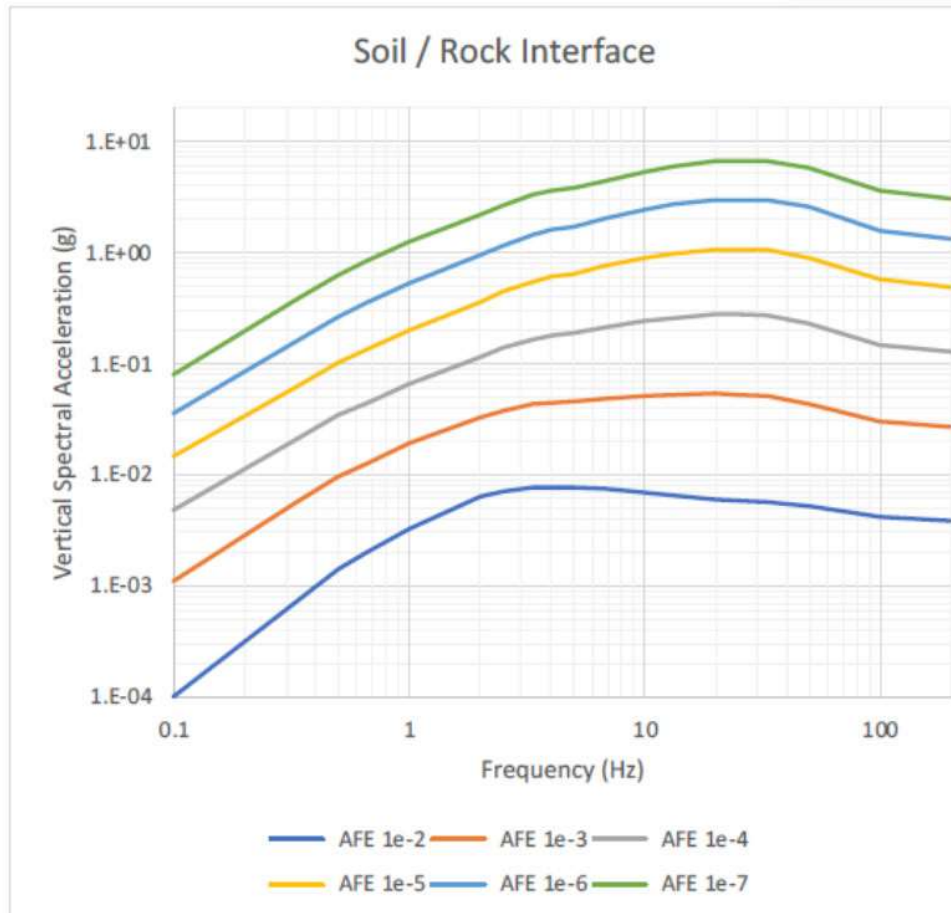




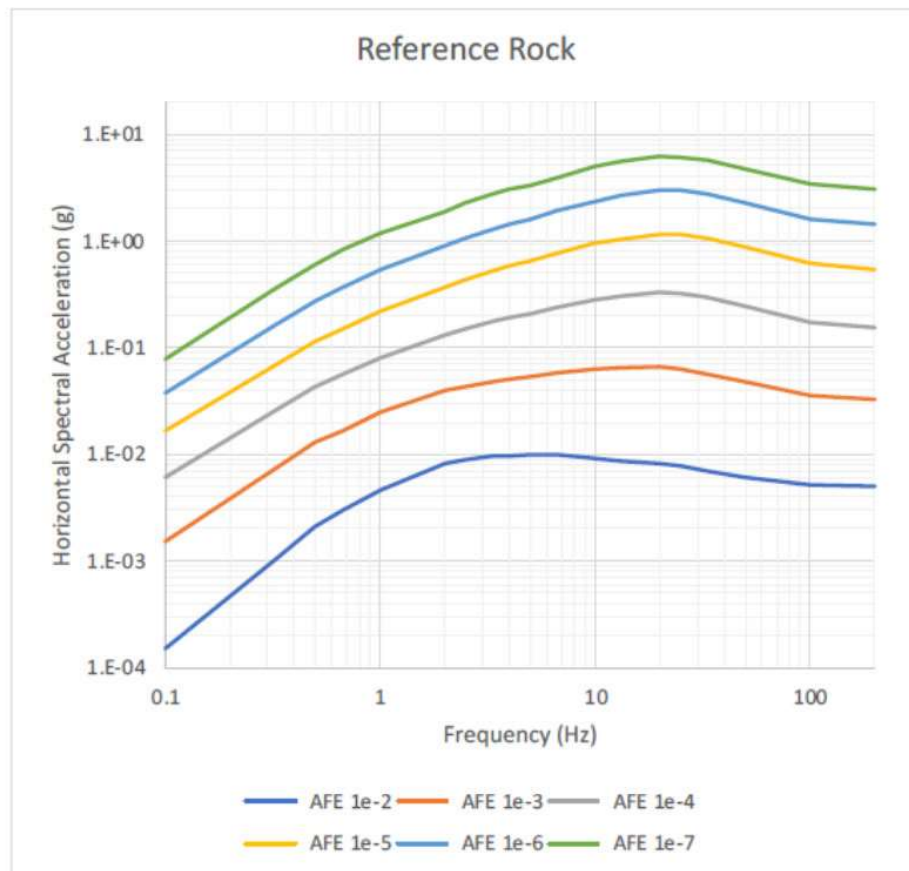
**Figure 2.7.4.6-2: Vertical UHRS at Elevation 52.93 m Based on Mean Hazard  
(Reference 2.7-41)**



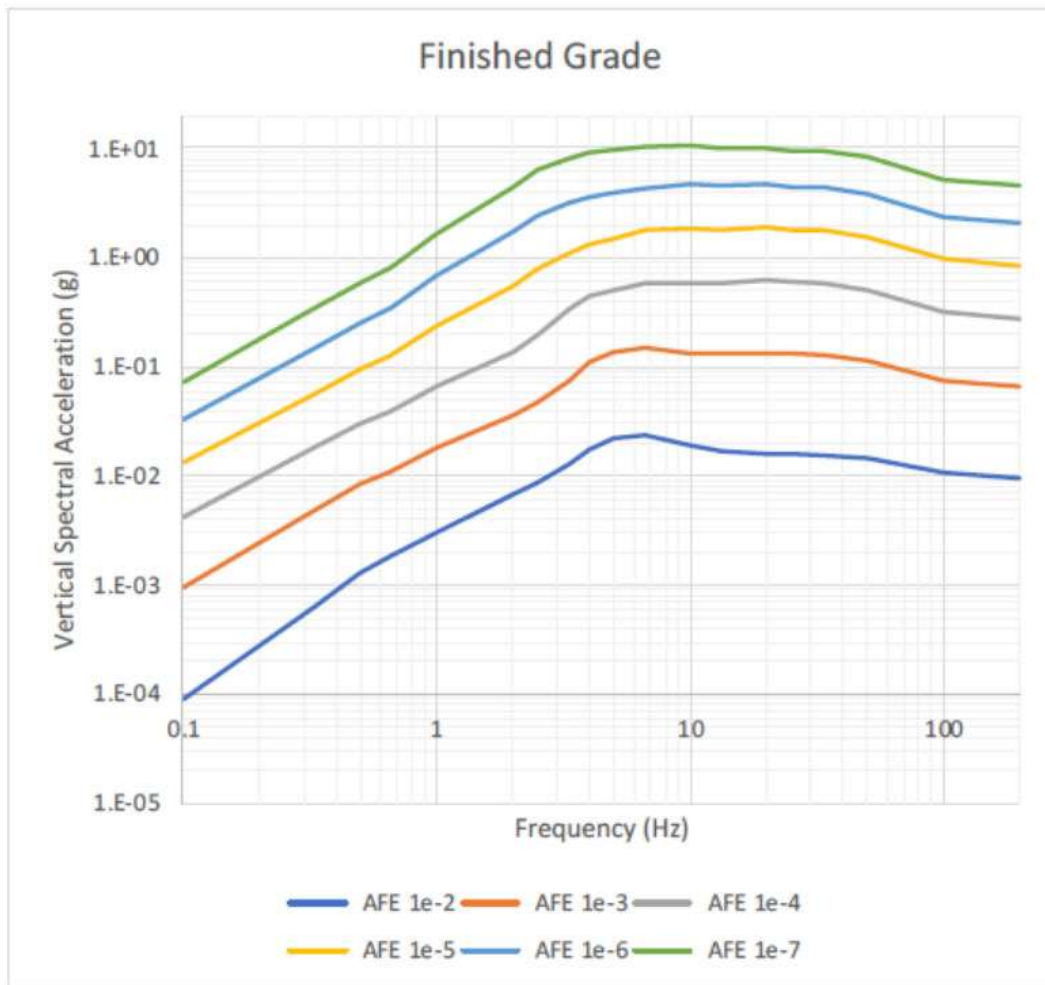
**Figure 2.7.4.6-3: Horizontal UHRS at Elevation 64 m Based on Mean Hazard  
(Reference 2.7-41)**



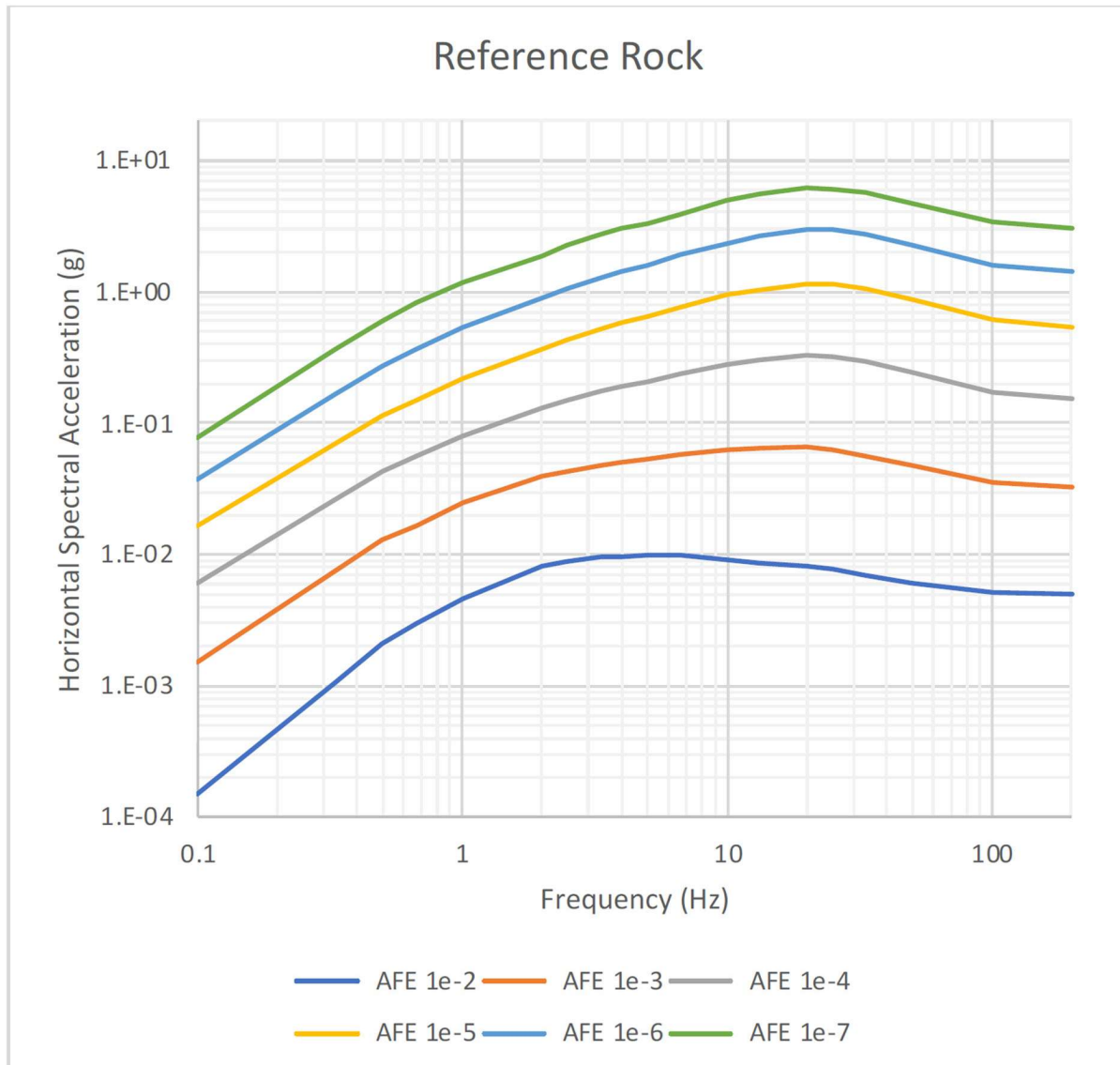
**Figure 2.7.4.6-4: Vertical UHRS at Elevation 64 m Based on Mean Hazard  
(Reference 2.7-41)**



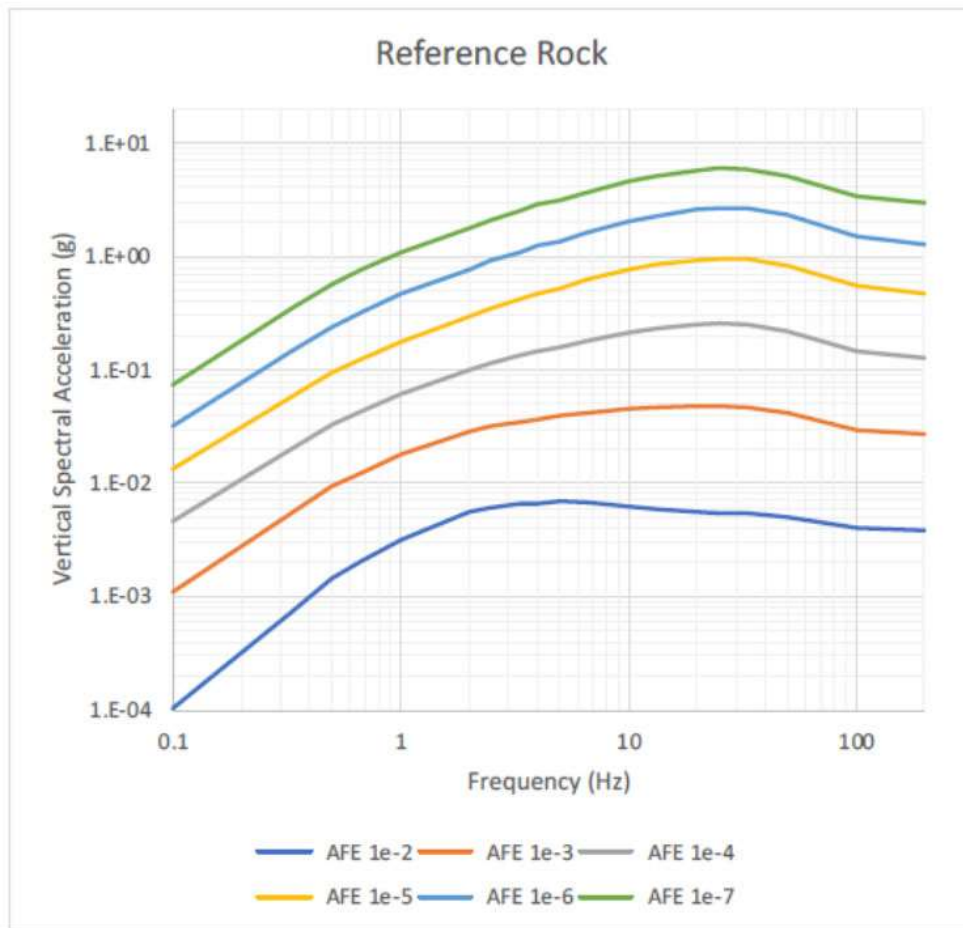
**Figure 2.7.4.6-5: Horizontal UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)**



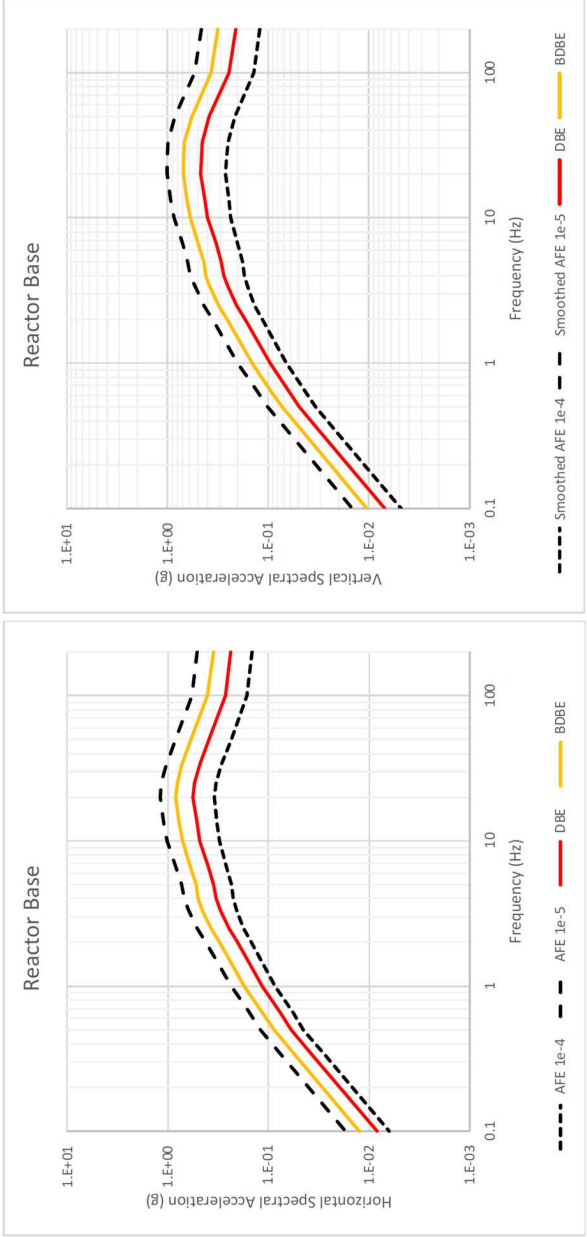
**Figure 2.7.4.6-6: Vertical UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)**



**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)**



**Figure 2.7.4.6-9: Initial Horizontal DBE and BDBE Spectra for Reactor Base (Elevation 52.93 m)  
(Reference 2.7-41)**



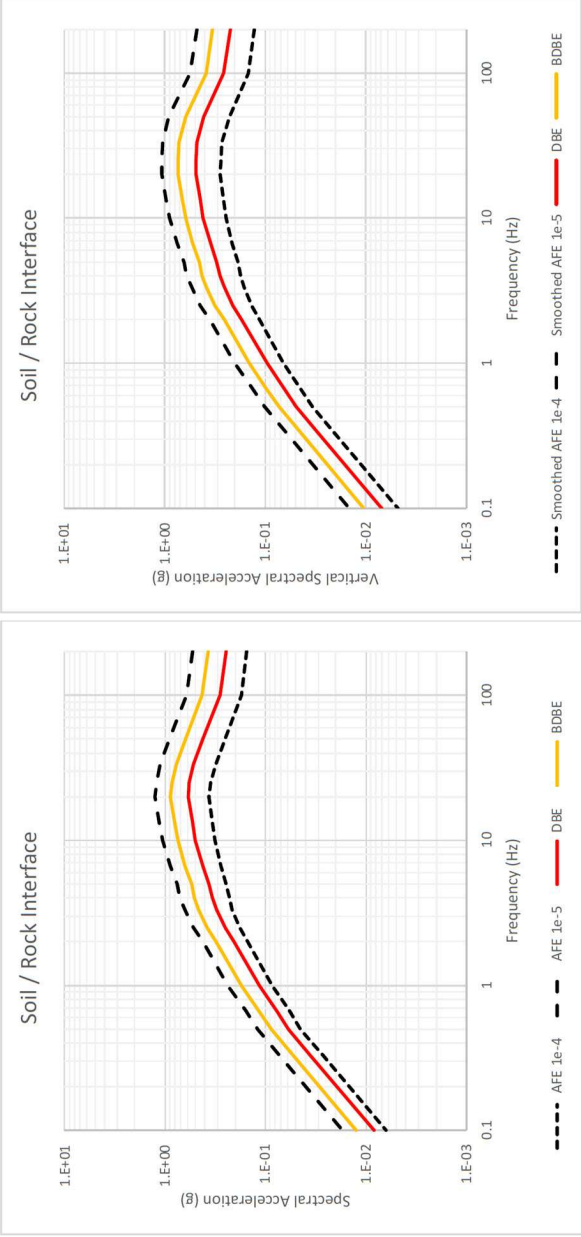


Figure 2.7.4.6-10: Initial Horizontal DBE and BD8E Spectra for Soil-Rock Interface (Elevation 64 m) (Reference 2.7.41)

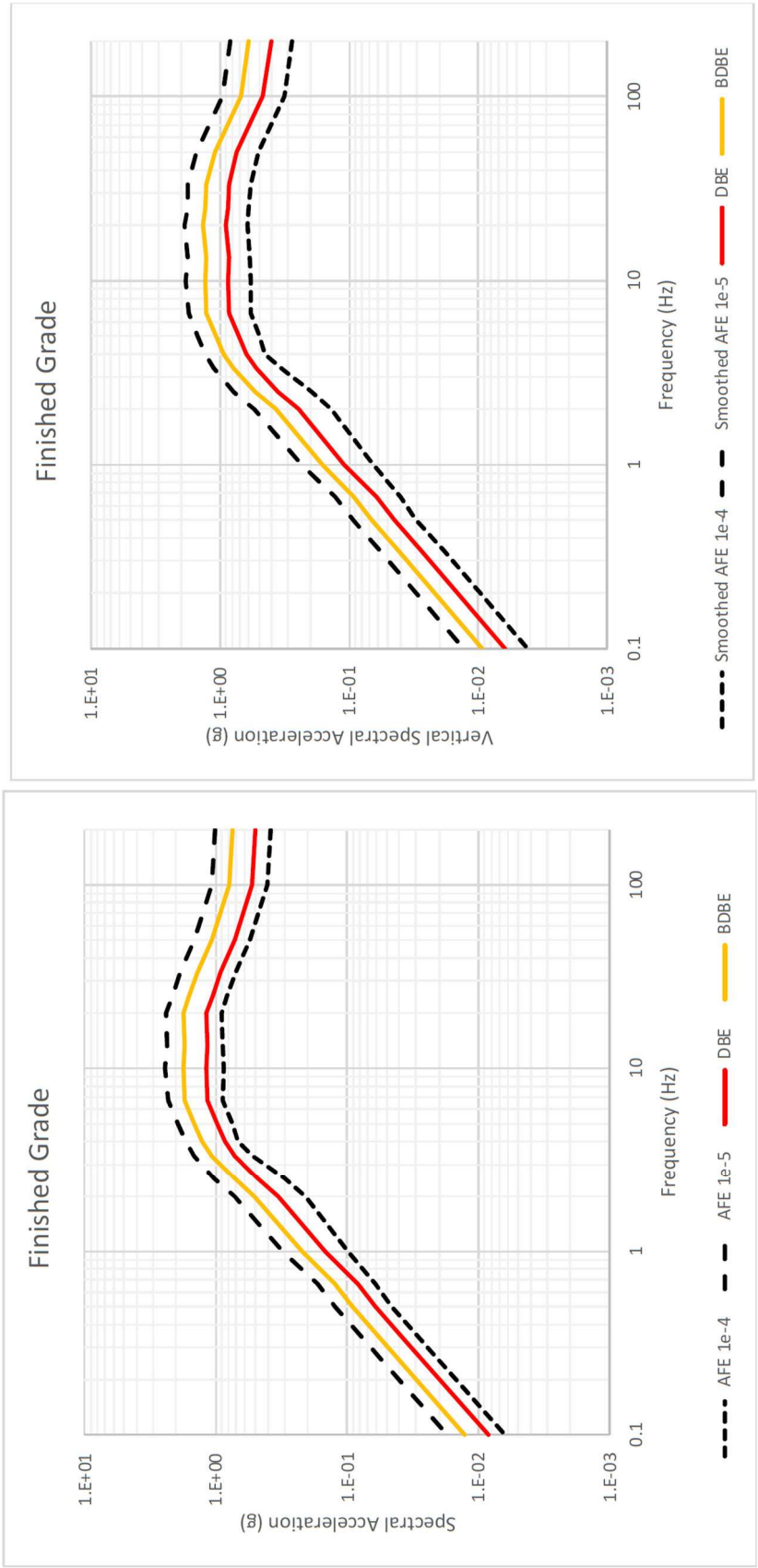


Figure 2.7.4.6-11: Initial Horizontal DBE and BDBE Spectra for Finished Grade (Elevation 88 m) (Reference 2.7-41)

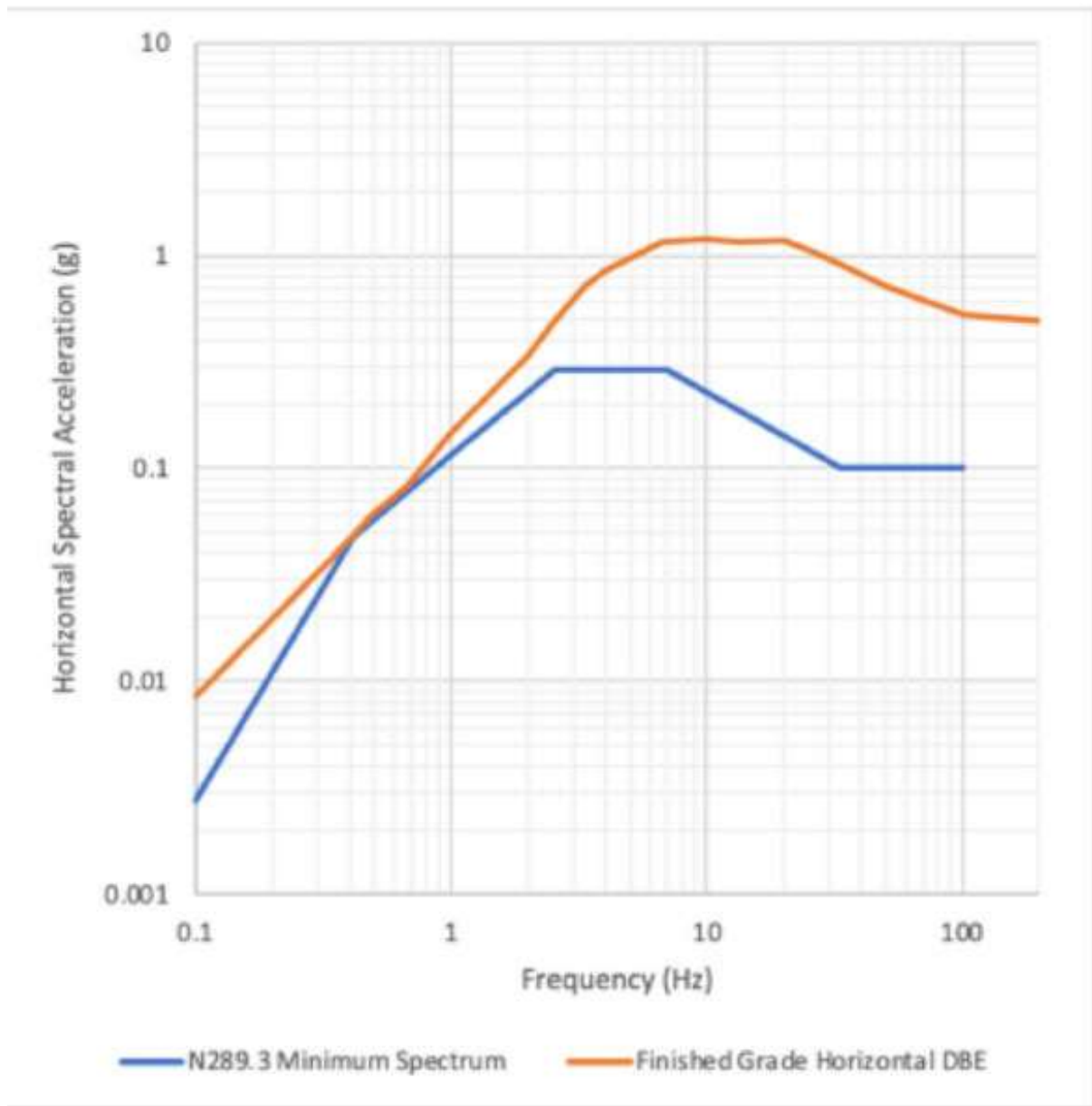


Figure 2.7.4.6-12: Comparison of Finished Grade Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)

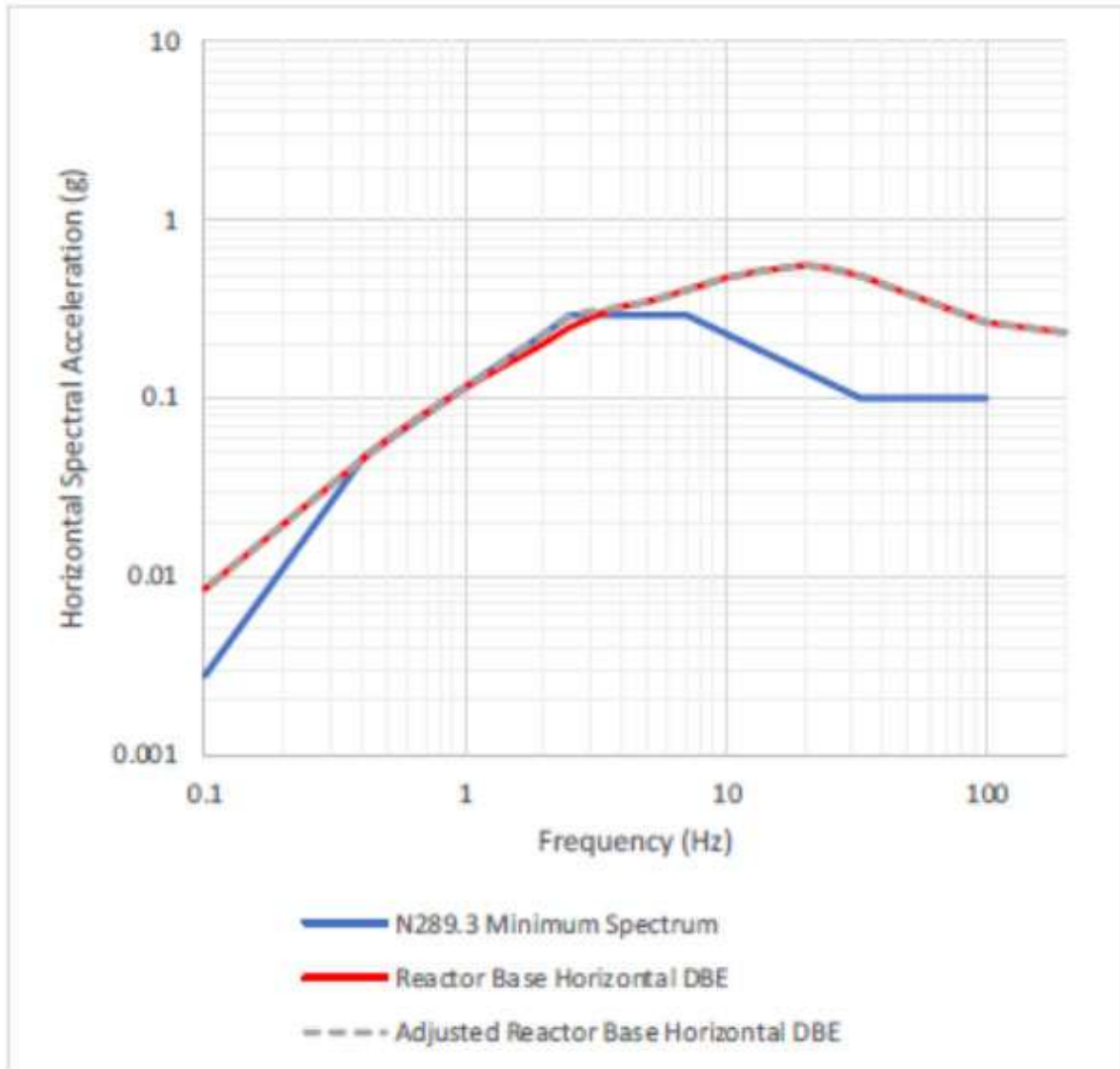


Figure 2.7.4.6-13: Comparison of Reactor Building Base Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)



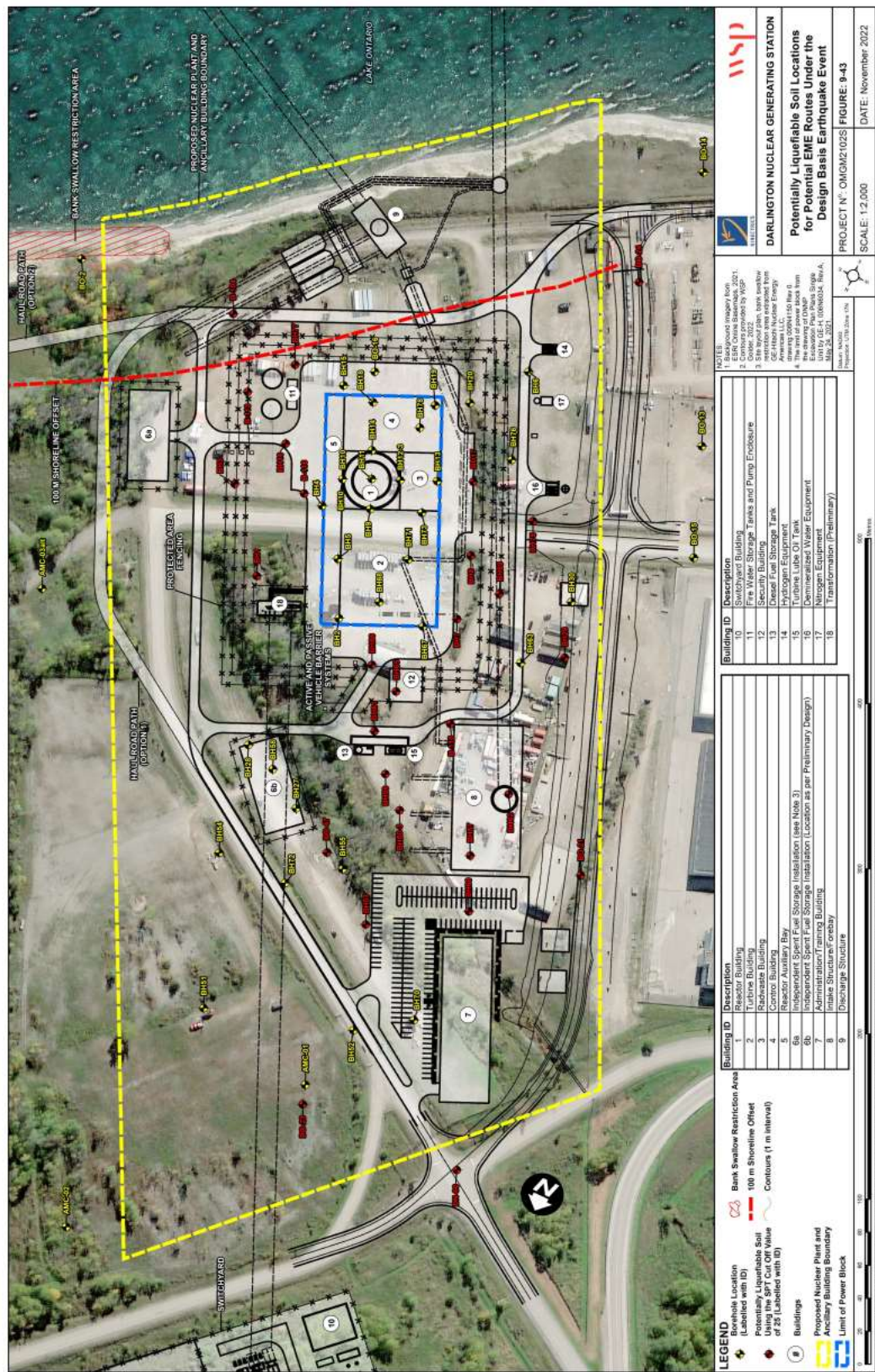


Figure 2.7.4.7-1: Locations of Boreholes Potentially to Liquefaction Under the DBE Event (Reference 2.7-42)



[illegible]

2-170

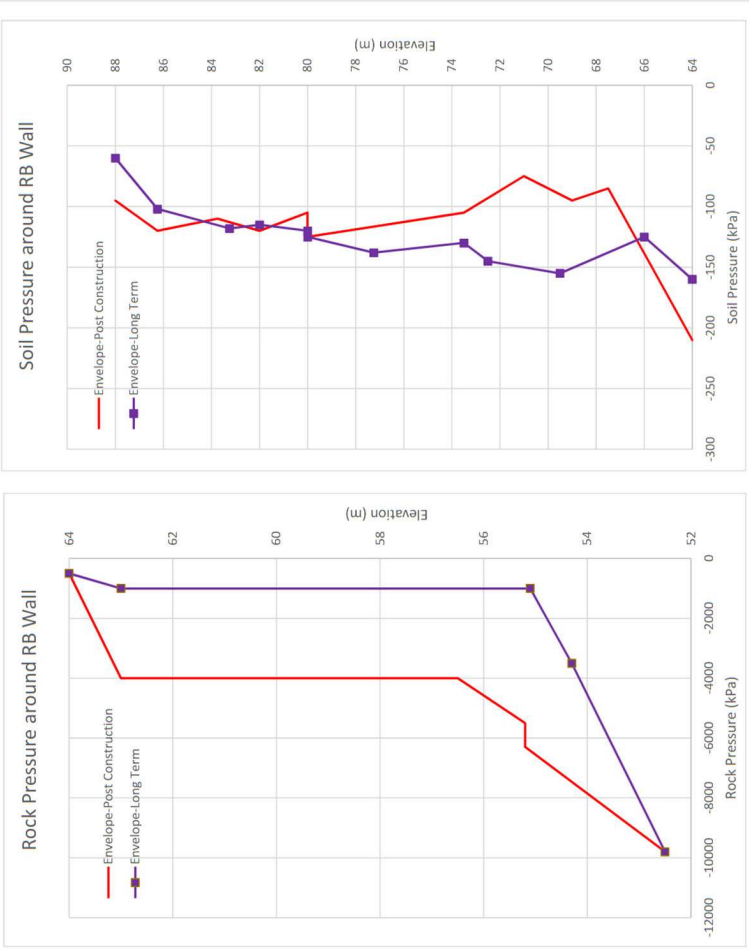


Figure 2.7.5.1.3-1: Rock and Soil Pressures Around RB Wall (Reference 2.7-38)

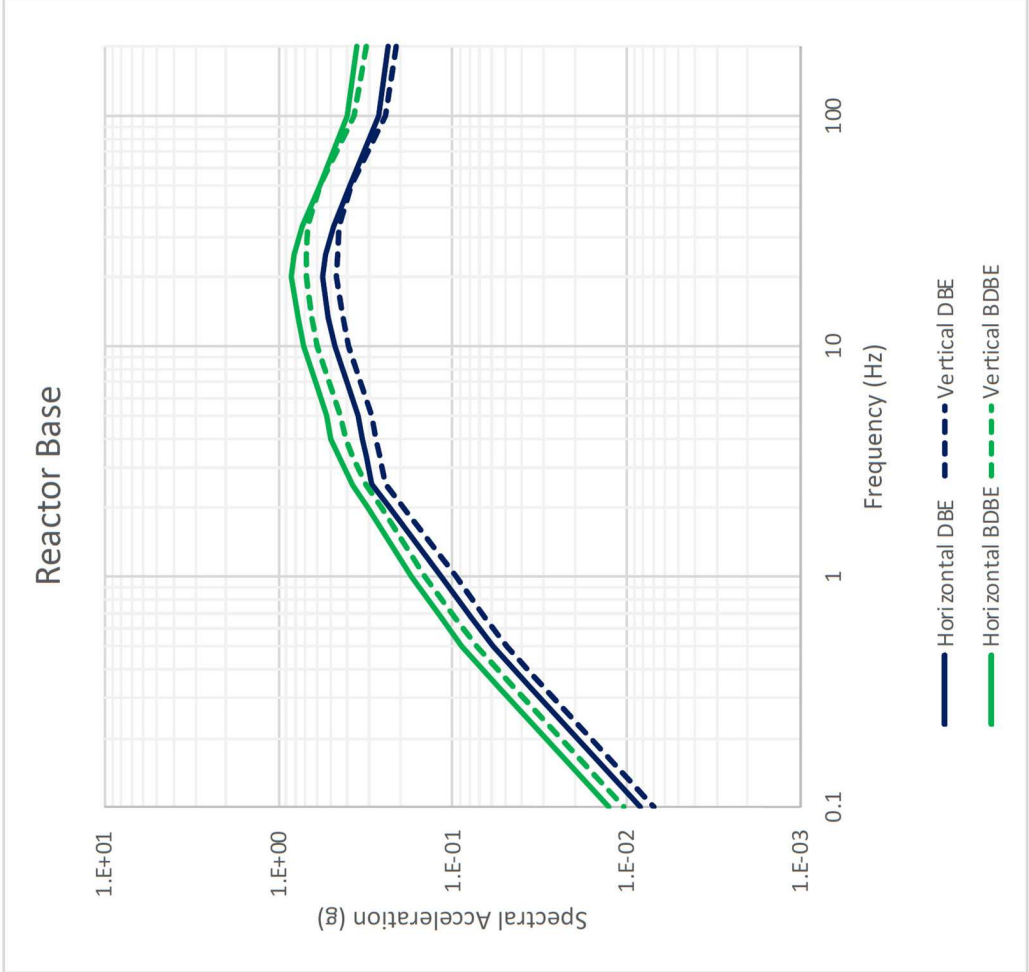
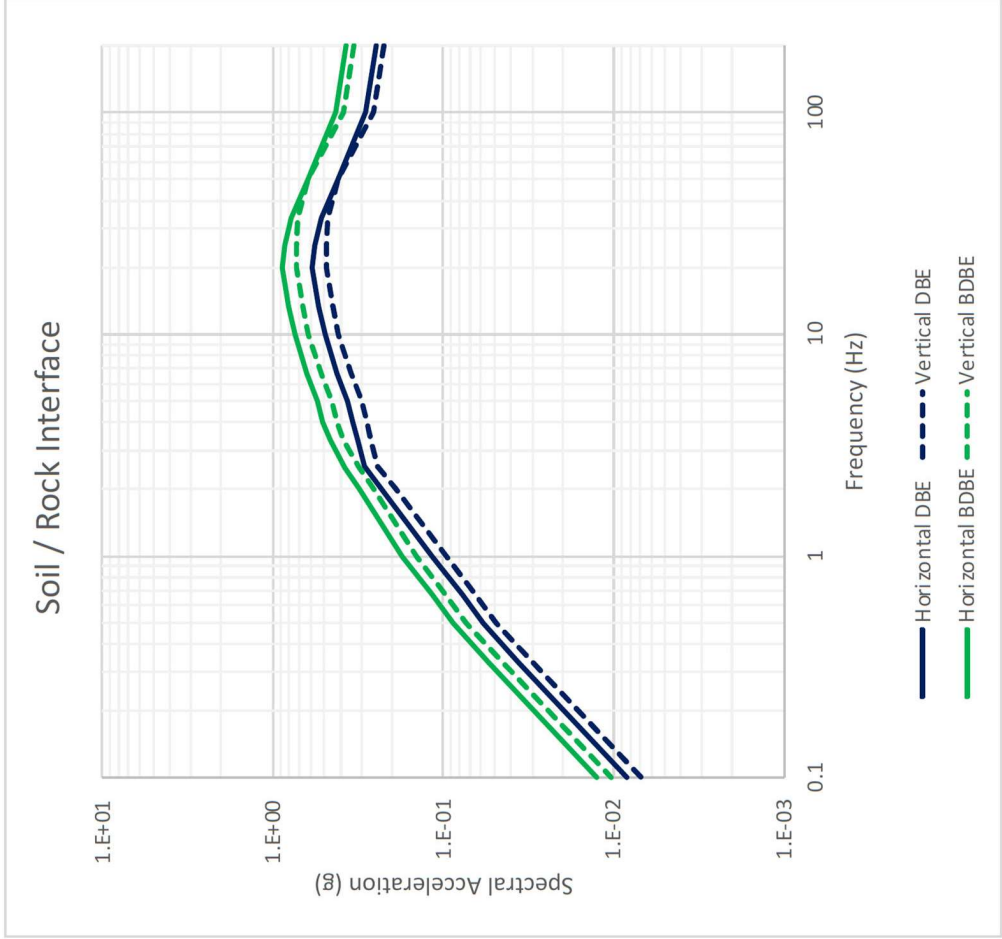


Figure 2.7.5.2.5-1: DBE and BDBE Foundation Input Response Spectra for Reactor Base (Elevation 52.93 m) (Reference 2.7-41)





**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)**

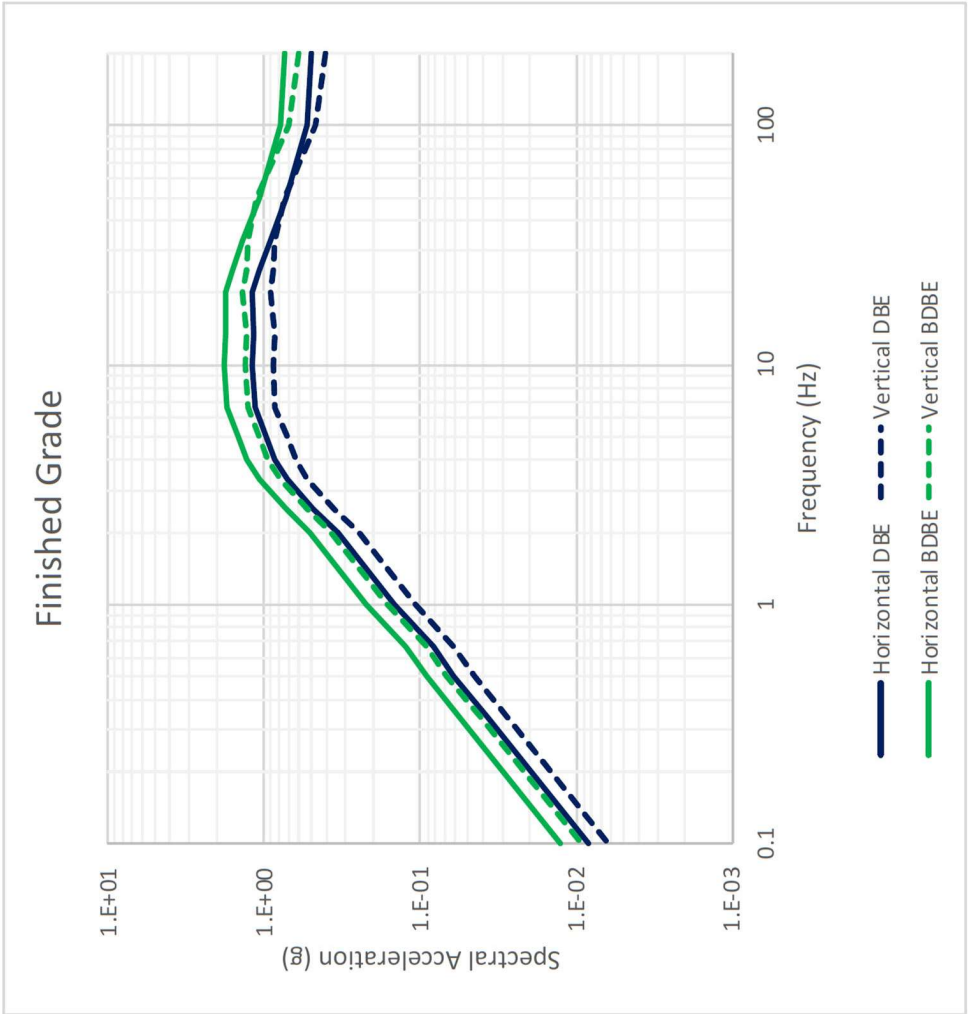


Figure 2.7.5.2.5-3: DBE and BDBE Performance Based Surface Response Spectra for Finished Grade (Elevation 88 m)  
(Reference 2.7-41)

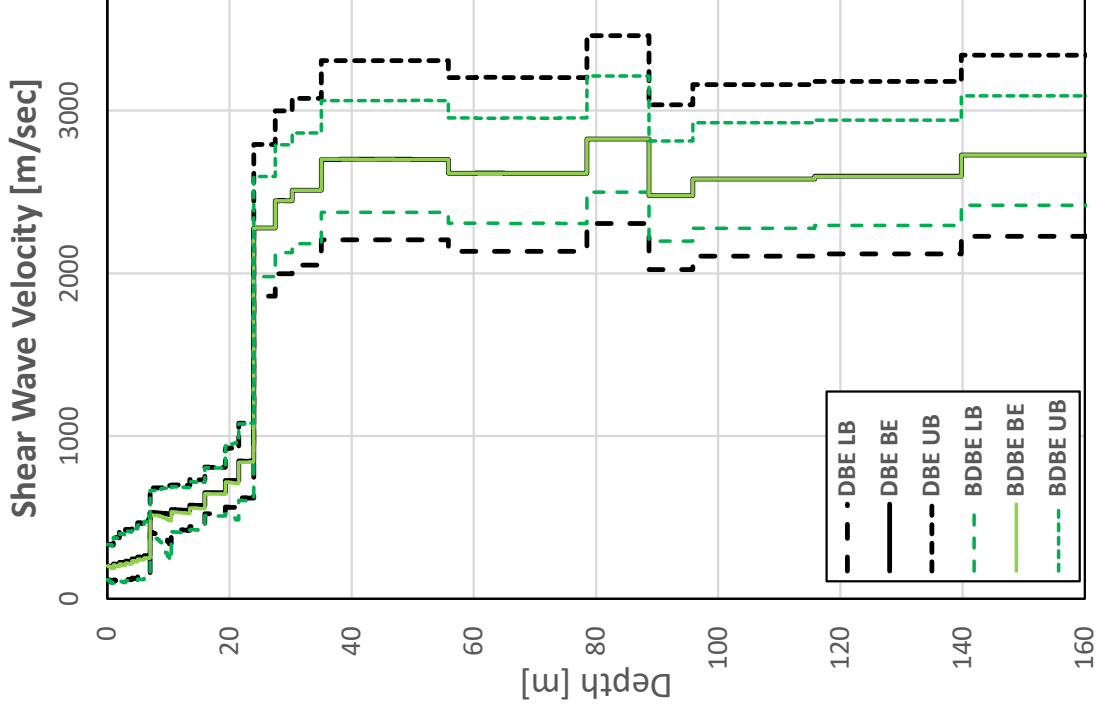


Figure 2.7.5.2.5-4: Subgrade Profiles of DBE and BDBE HCSC Shear Wave Velocities (Reference 2.7.41)

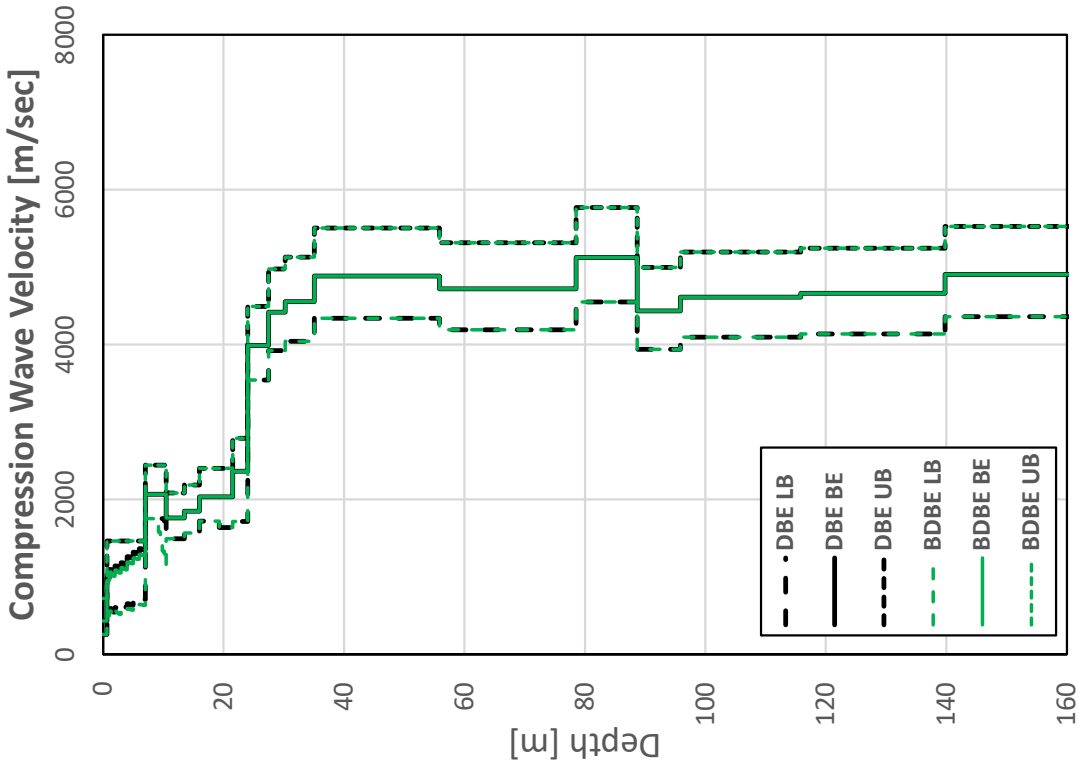


Figure 2.7.5.2.5-5: Subgrade Profiles of DBE and BDBE HCSC Compression Wave Velocities (Reference 2.7-41)

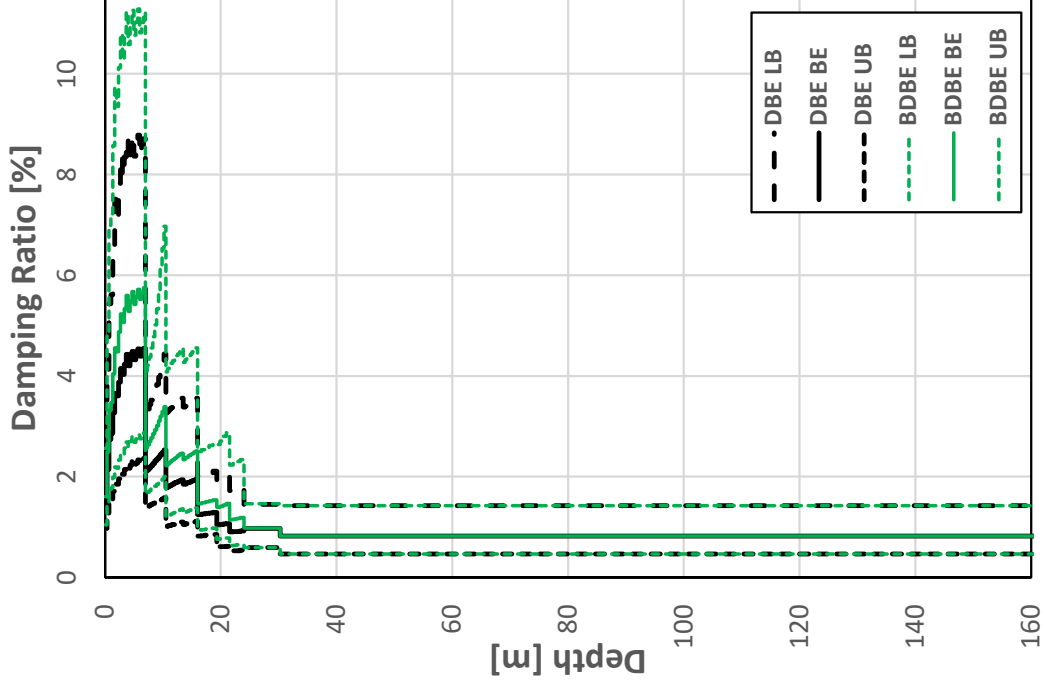


Figure 2.7.5.2.5-6: Subgrade Profiles of DBE and BDBE HCSC Damping Ratios (Reference 2.7-41)

## **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.

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**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			

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**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%)



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**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 (Based on Site-specific Survey (2018) and Pathway Analyses (2016))</b>		
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
	<p><b>2.8.6 Impact of Biological Data</b></p> <p>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)</p>

### **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).

**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).

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**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	-	-	-

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### 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)**

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	<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

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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.

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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).



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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).

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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.

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**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>		

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**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>NOTES:</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	
<b>2.9.2.1.2</b> <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.
<b>2.9.2.1.3</b> <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



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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.

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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

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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:

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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.

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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.



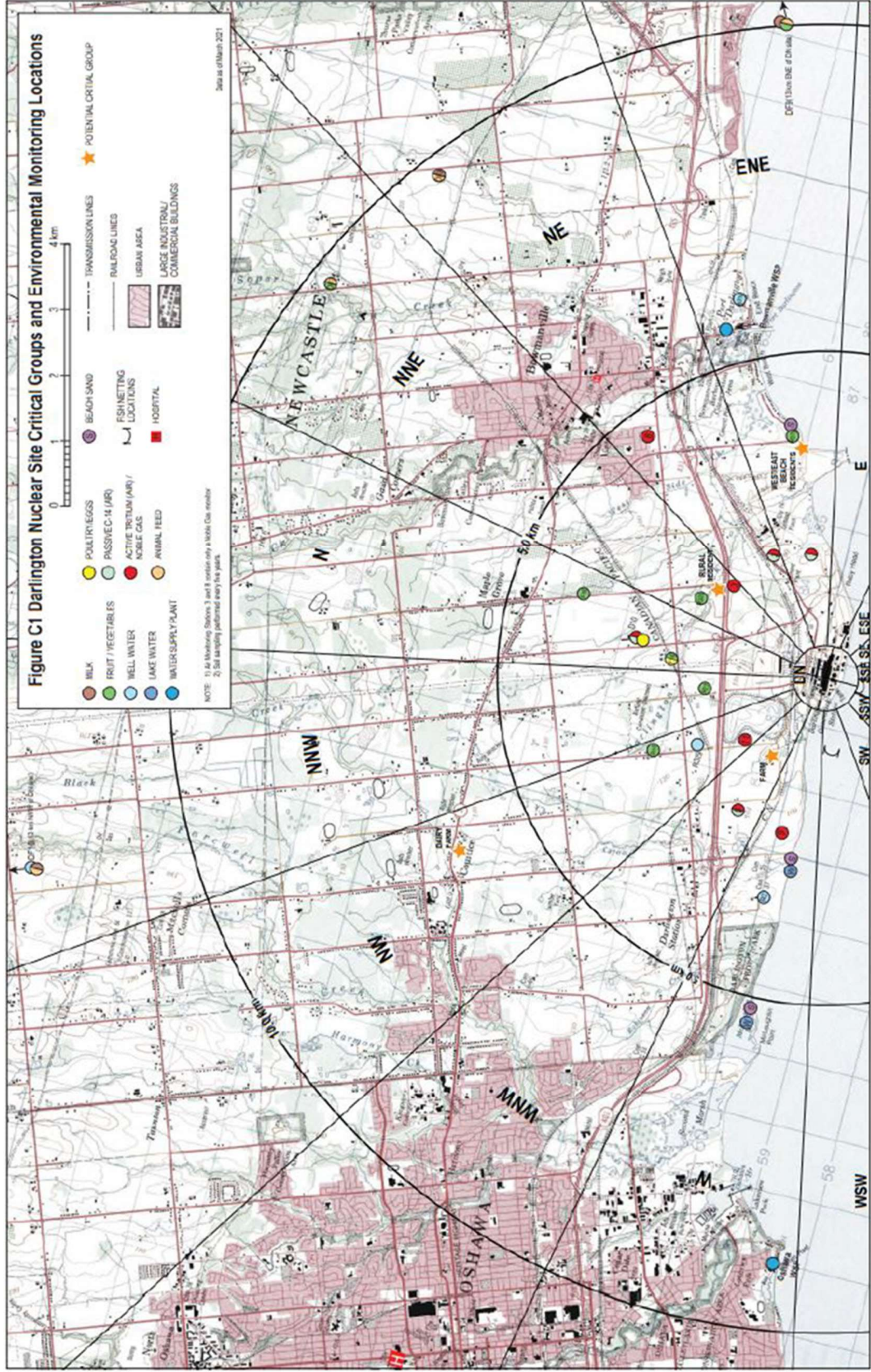


Figure 2.9-1: Darlington Nuclear Site Critical Groups and Environmental Monitoring Locations (Reference 2.9-2)



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**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, "DNNP – 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).

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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.

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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

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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.

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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).



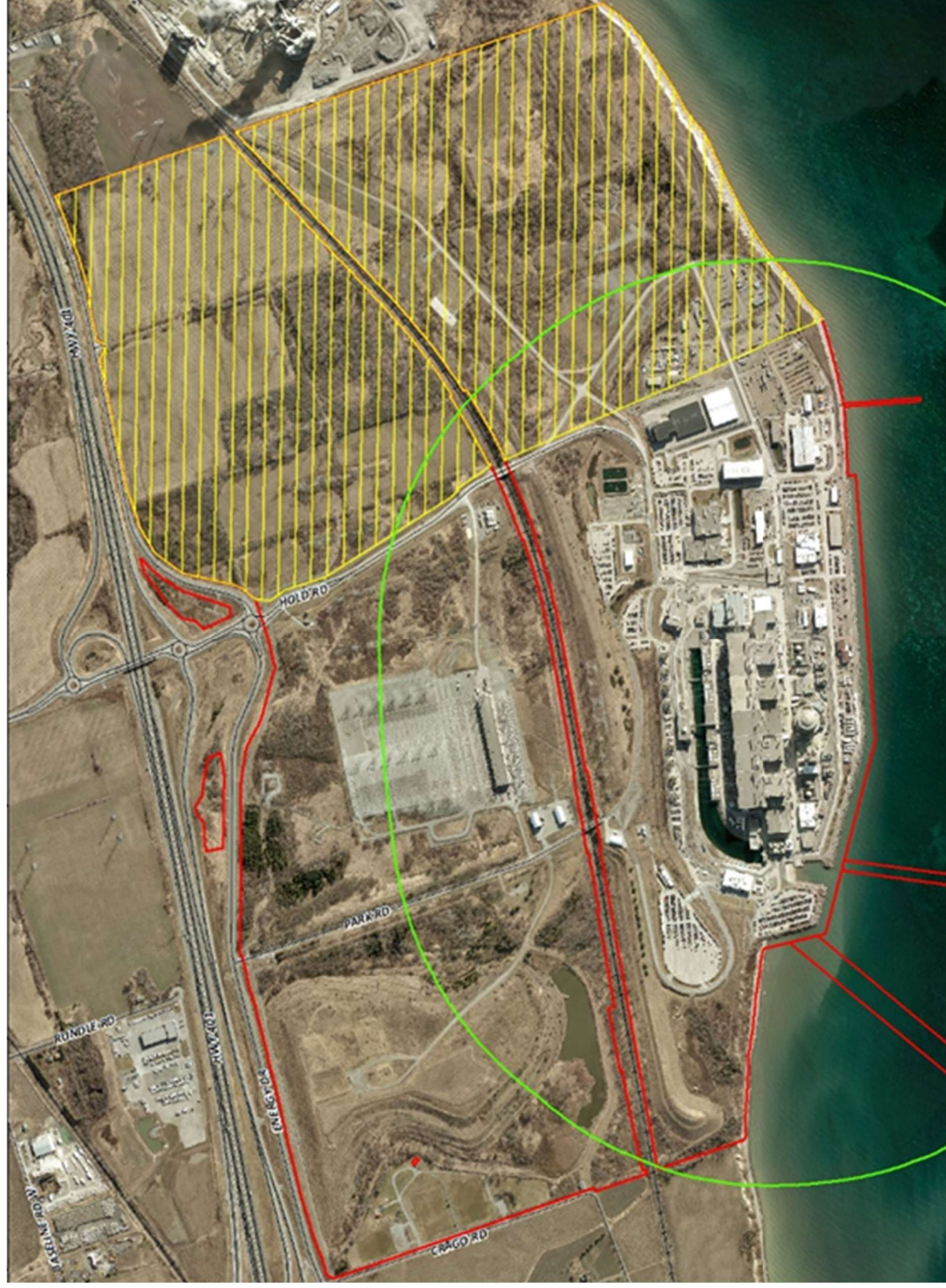


Figure 2.10.2-1: Darlington Nuclear Site Showing DNGS and DNNP Areas



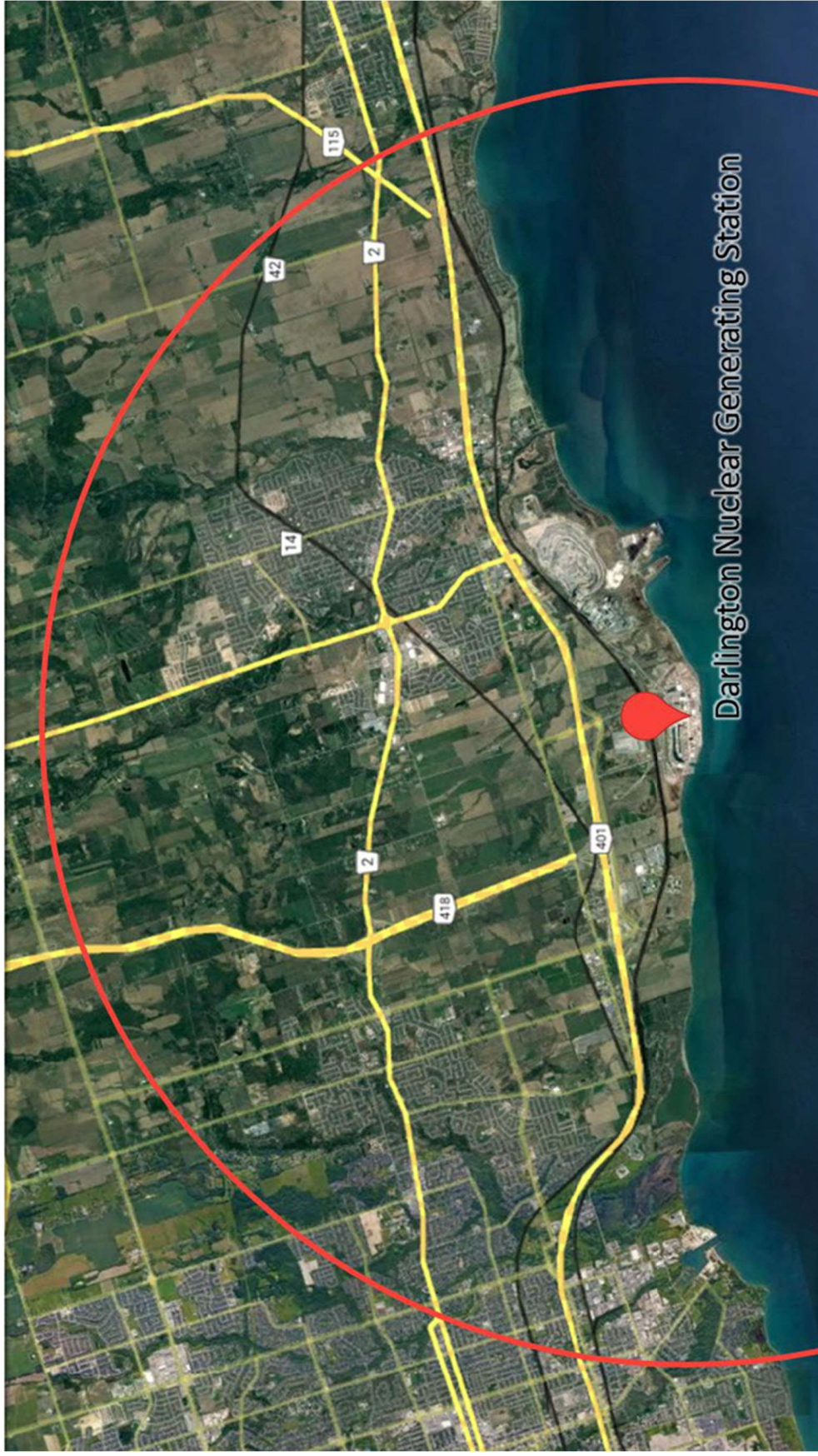


Figure 2.10.2-2: Area of Consideration for Traffic Management Plan

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**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

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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.

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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-



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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

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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



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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**

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- 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.

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- 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.

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.2	Foundation Interface Analysis (FIA) <u>Status:</u> Complete	<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>

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.3	<p>Site Geotechnical and Seismic Hazard Investigation Plan</p> <p><u>Status:</u> Complete</p>	<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>

**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>

**Table 2.12-1: DNNP Projects Closure Plans and Associated Updates**

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.5	Climate Change Impact  <u>Status:</u> “CNCS Deliverable 1: DNNP Strategy for Addressing Climate Change Impacts.”  Complete	<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	3-second Wind Gust Calculation  <u>Status:</u> Complete	<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>

**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	<p>Work started to finalize appropriate consideration for snow load with a Winter Probable Maximum Precipitation (PMP) event. DNNP considers this a review level condition.</p> <p>Finalization of the coincident snow load and winter PMP is complete.</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. NK054-REP-02730-00004 R000, 2022, "Winter PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-004 R01)</li> </ol>
2.12.8	PMP Validation <u>Status:</u> Complete	<p>Confirmation of rainfall and PMP</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> <li>1. NK054-REP-02730-00002 R000, 2022, "PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-002 R01)</li> </ol>



## **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.

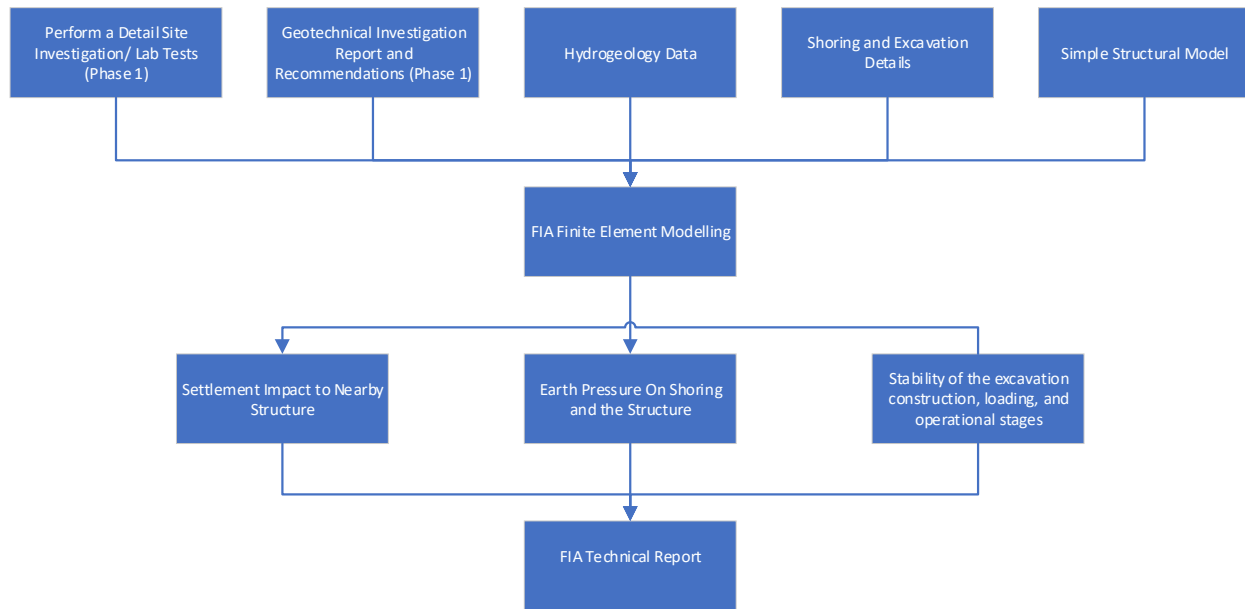
The Technical Report is prepared based on the FIA modelling, includes the results of the FIA of the deeply embedded BWRX-300 RB and the surrounding Power Block foundations at the DNNP site. The report discusses:

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.

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**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**

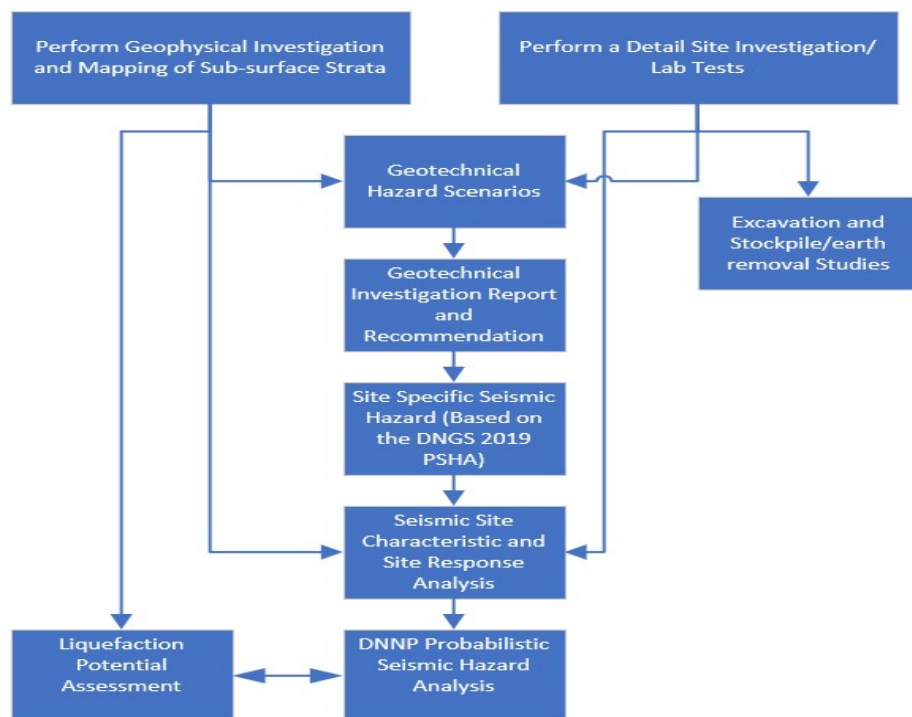
1. Completed geophysical investigation and mapping of subsurface strata
2. Completed detailed site investigation and laboratory tests
3. Drafted report on the geophysical investigation and laboratory tests as well as recommendations
4. Excavation and earth removal studies continue
5. Site-specific characteristics 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

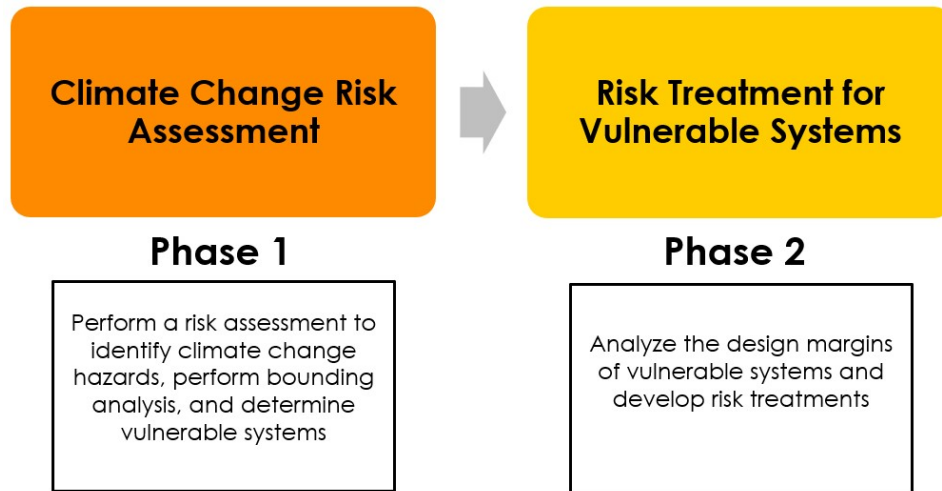
Work will be performed on an as required basis to integrate climate change assessments into the current nuclear safety framework.

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



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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

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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

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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

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**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.

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**APPENDIX B – List of Roads within the Survey Area**

Name of Road / Highway / Station	Direction	Road Type
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

Name of Road / Highway / Station	Direction	Road Type
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

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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



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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

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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
Clarrington 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

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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

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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

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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
Thorncliffe 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

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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

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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

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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



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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

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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

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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

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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

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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
Mossgrove 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

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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

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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

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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



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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  
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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

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<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

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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.



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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.

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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.	

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**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

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<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

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<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

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<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

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<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

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### 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.

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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  $1\text{E-}5$  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  $1\text{E}15$  becquerels of Iodine-131, shall be less than  $1\text{E-}5$  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  $1\text{E}14$  becquerels of Cesium-137 shall be less than  $1\text{E-}6$  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):

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- **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. DECs are a subset of beyond-design-basis accidents (BDBA), and are therefore, not part of the design basis. DECs are considered in the design process of the facility in accordance with best-estimate methodology DECs 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 DECs 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 DECs 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 DECs 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.)

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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:

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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



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- 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  $1E-05$  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 DECs 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.

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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 DECIs, 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.

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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,

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- 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

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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."

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- 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.

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**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



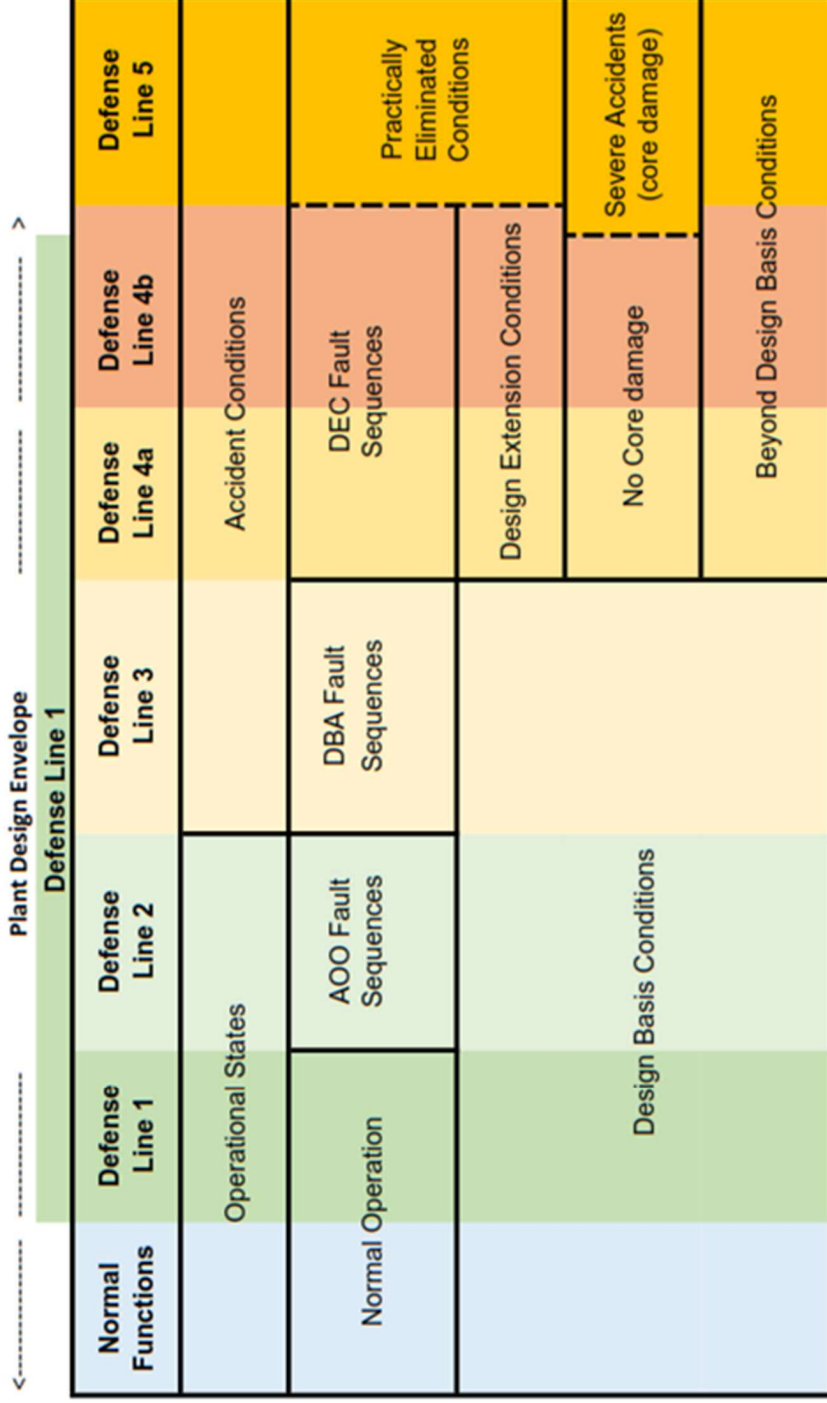


Figure 3.1-1: Defence-in-Depth – Plant States and Defense lines

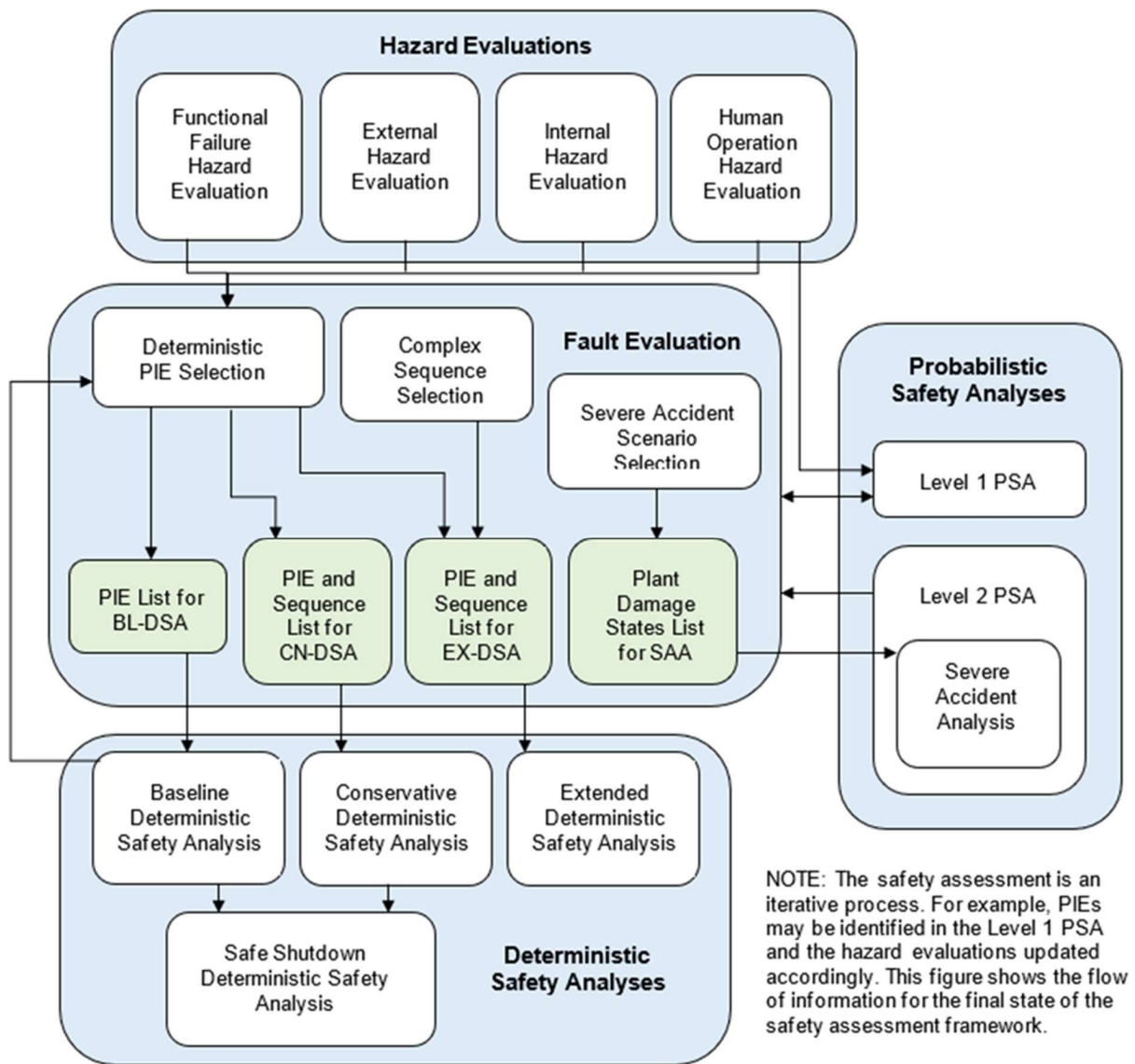


Figure 3.1-2: BWRX-300 Safety Strategy Implementation Process

### **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.

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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).

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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),



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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.

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- 3.2-19 Canadian Commission on Building and Fire Codes, "National Building Code of Canada,"  
National Resource Council of Canada.

**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>

**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

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- 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

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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.



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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

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- 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).

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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 N60 values. Two correlations are used to estimate  $V_s$  for the N60 values, per the 2012 PEER Report 2012/08 (Reference 3.3-17). The selected  $V_s$  correlations use the N60 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-dependence 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

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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.

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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).

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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

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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.

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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.



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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

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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

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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

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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:

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- 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

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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:

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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

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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.

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- 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.

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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



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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.

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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

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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.

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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.

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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.

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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

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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:

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- 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)
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- 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.
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- 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.)
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- 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."
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- 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.
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- 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.

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**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

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**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

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**Table 3.3-3: Base Case Engineered Fill and In-situ Soil Dynamic**

Layer	Shear Wave Velocity (m/s)			Poisson's Ratio
	Base Case $V_s$	$\sigma_{\mu \text{ In}}$	$\sigma_{\mu \text{ In} V_s}$	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



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**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

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**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

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**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

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**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

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**Table 3.3-9: Seismic Damping Values for Piping and Equipment**

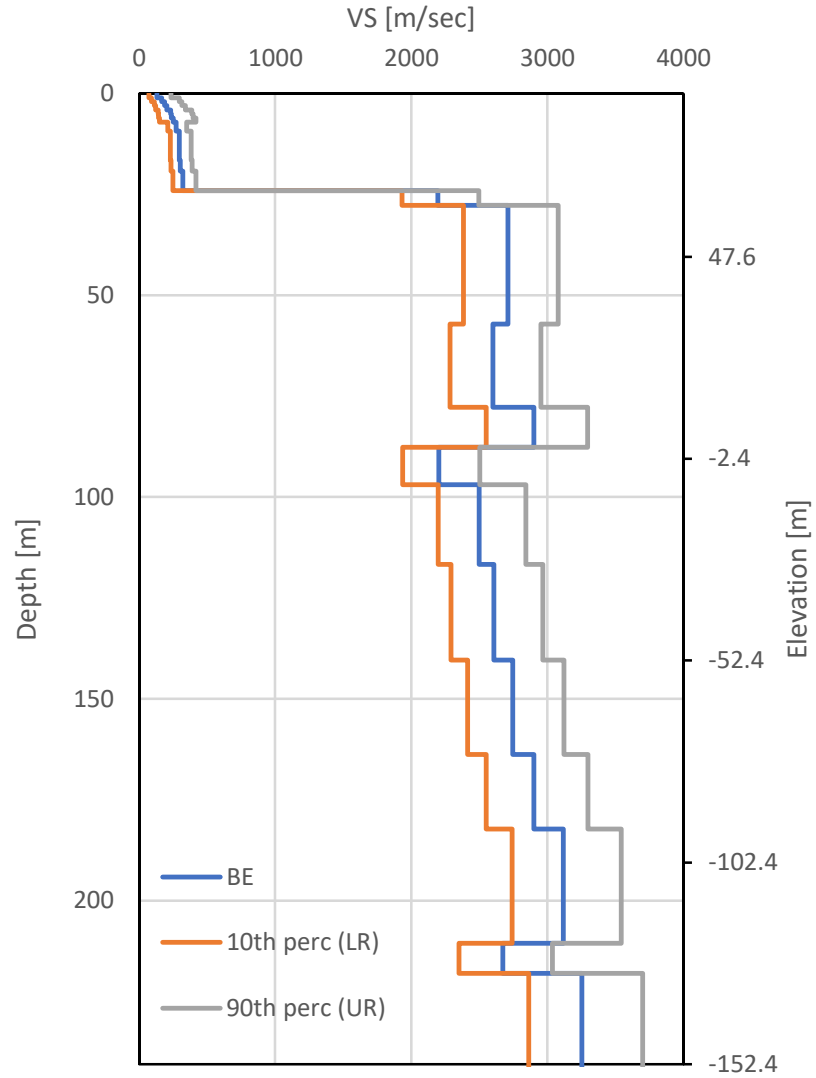
Structure or Component	Level 1 Damping	Level 2 Damping
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	3	5
Small-diameter piping systems, diameter equal to or less than 12 in.	2	5
Welded steel structures	3	4
Bolted steel structures	4	7

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**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

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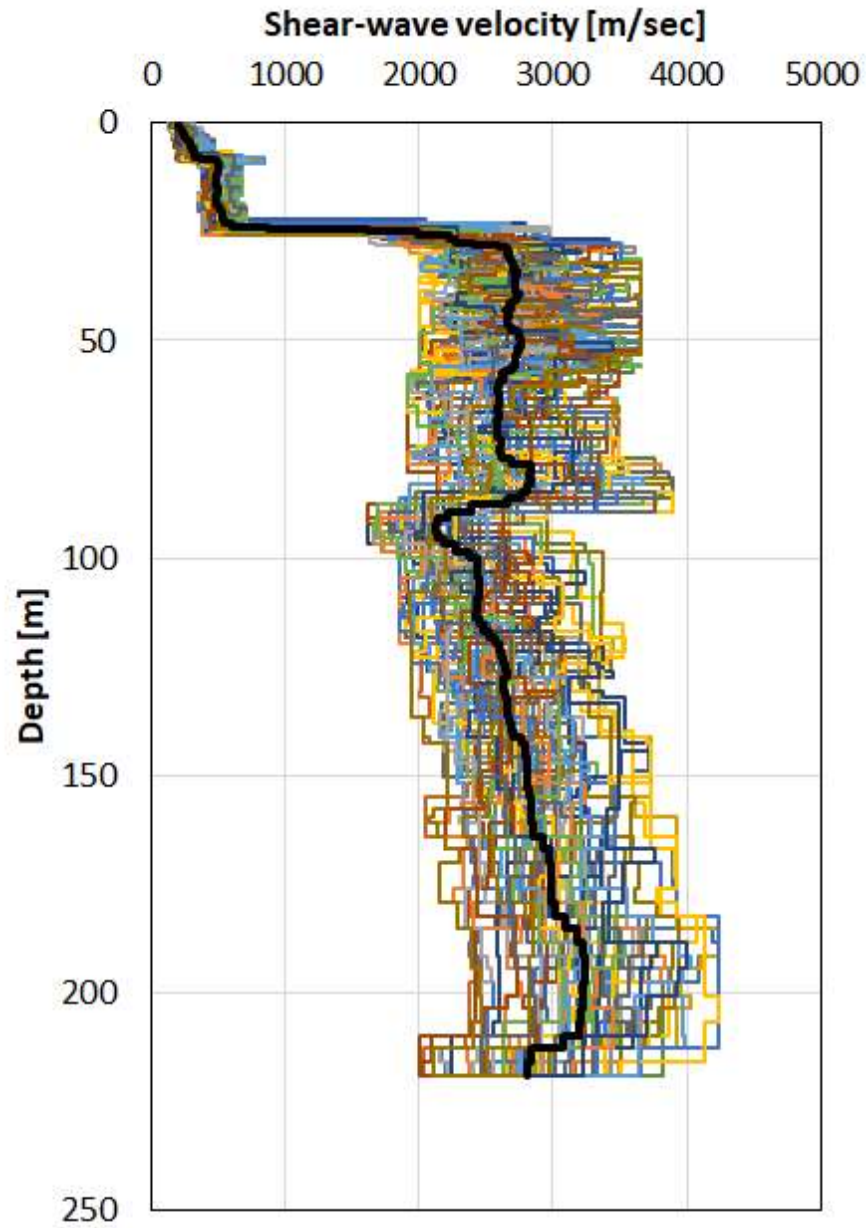


**Figure 3.3-1: Shear Wave Velocities for the Bounding In-situ Profile**



<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
		$\Sigma = 1.0$

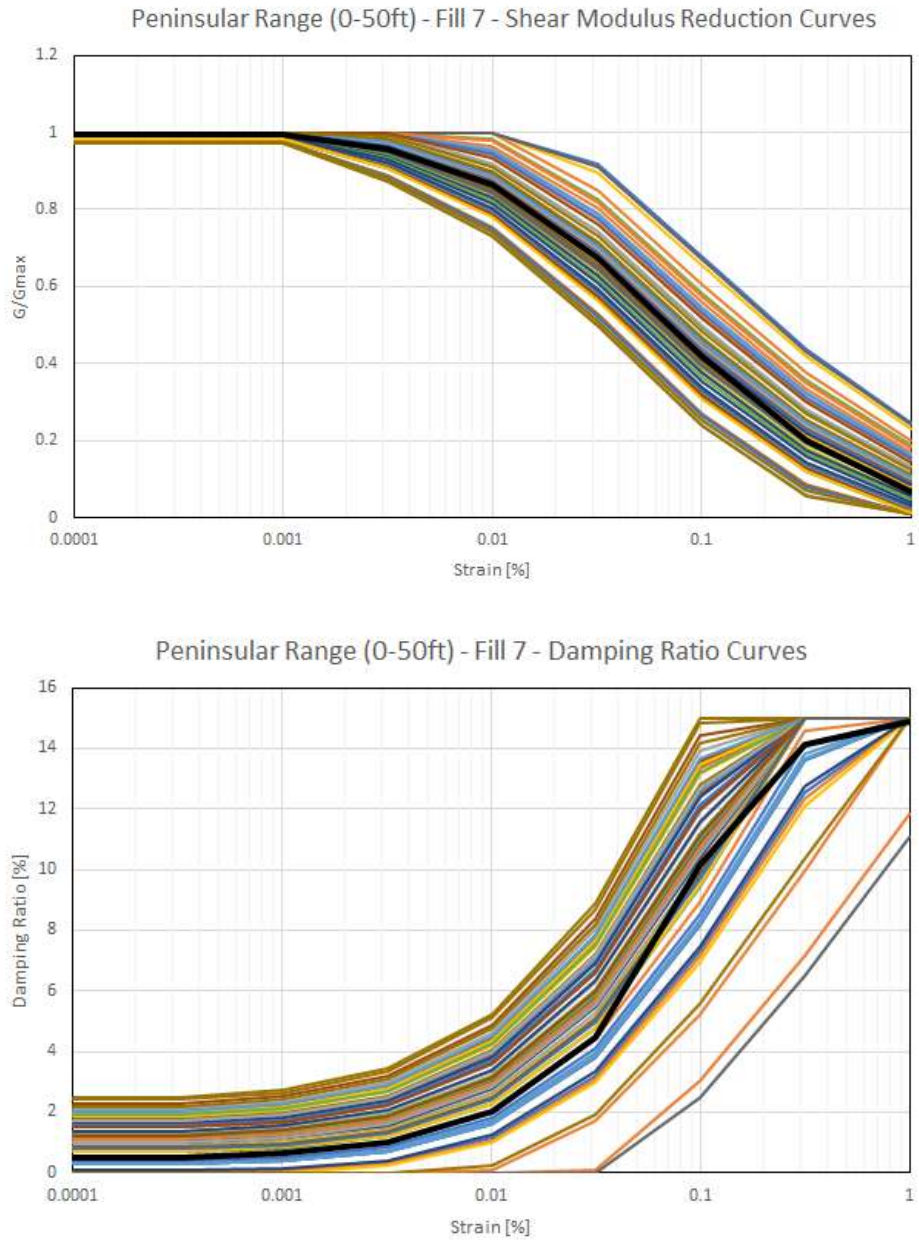
**Figure 3.3-2: Cases Considered for Explicit Considerations of Epistemic Uncertainties**



**Figure 3.3-3: Shear Wave Velocity and Layer Thicknesses Randomization – BE-BE Case**

Note: The Black line designates the resulting mean curve

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**Figure 3.3-4: Soil Degradation Curves Randomization**

Note: The Black line designates the resulting mean curve

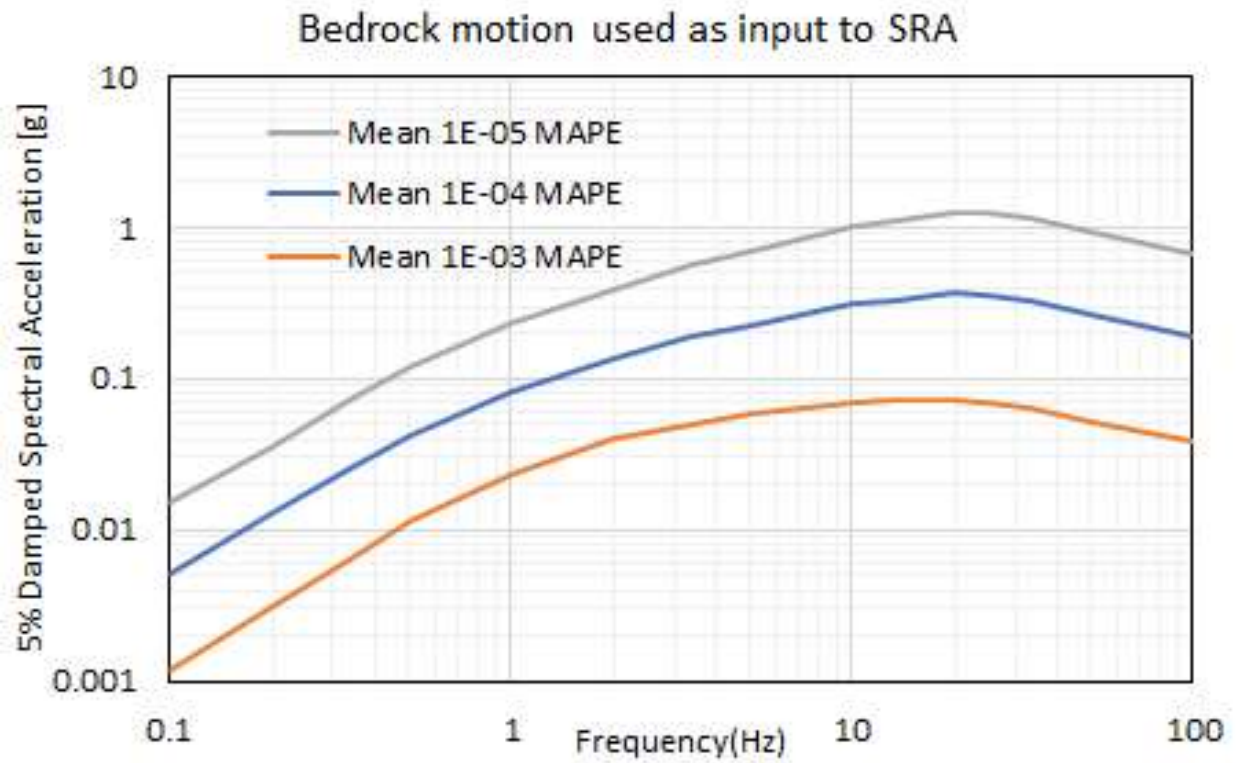


Figure 3.3-5: Uniform Hazard Response Spectra at Bedrock Elevation

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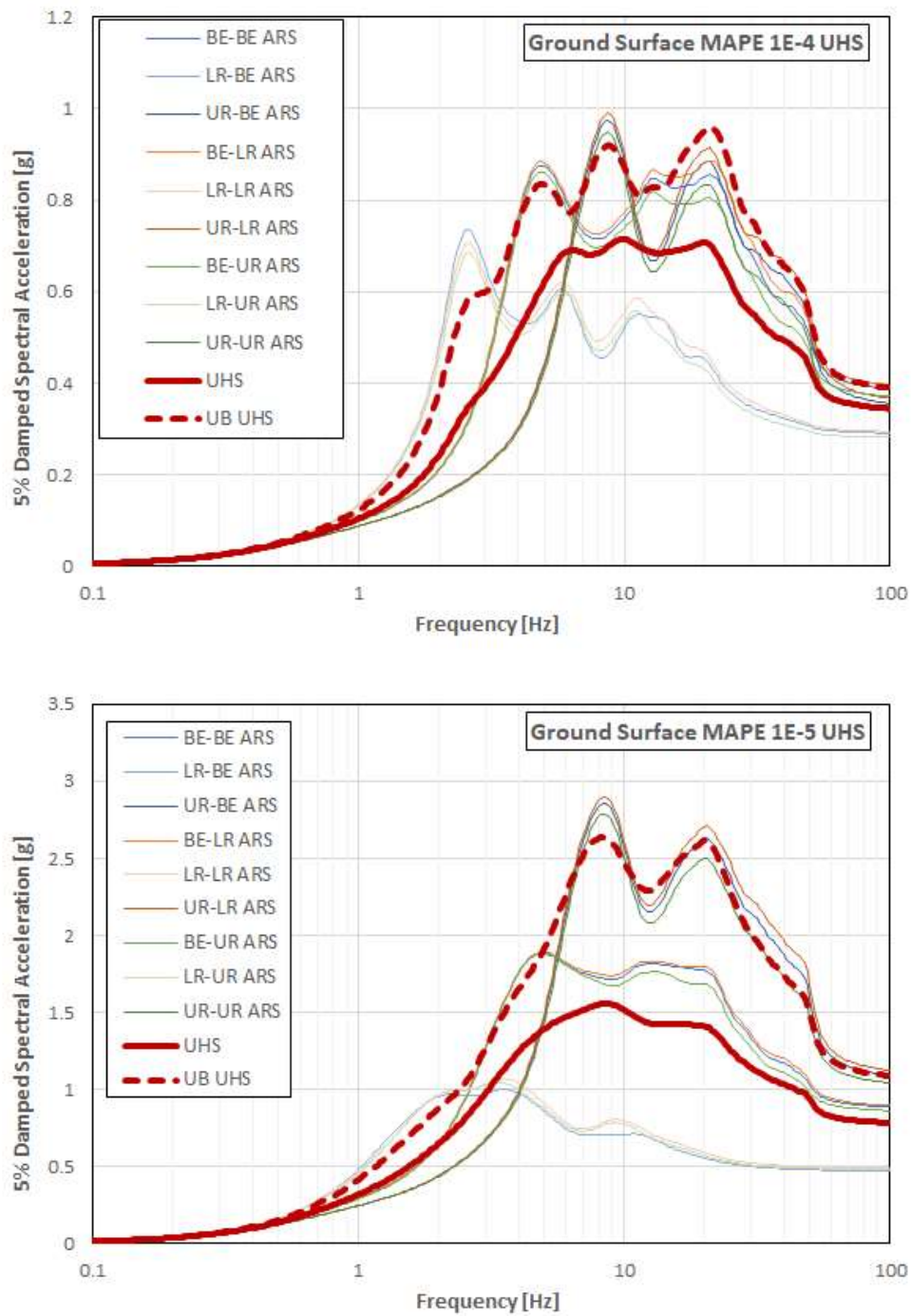


Figure 3.3-6: Ground Surface Uniform Hazard Response Spectra

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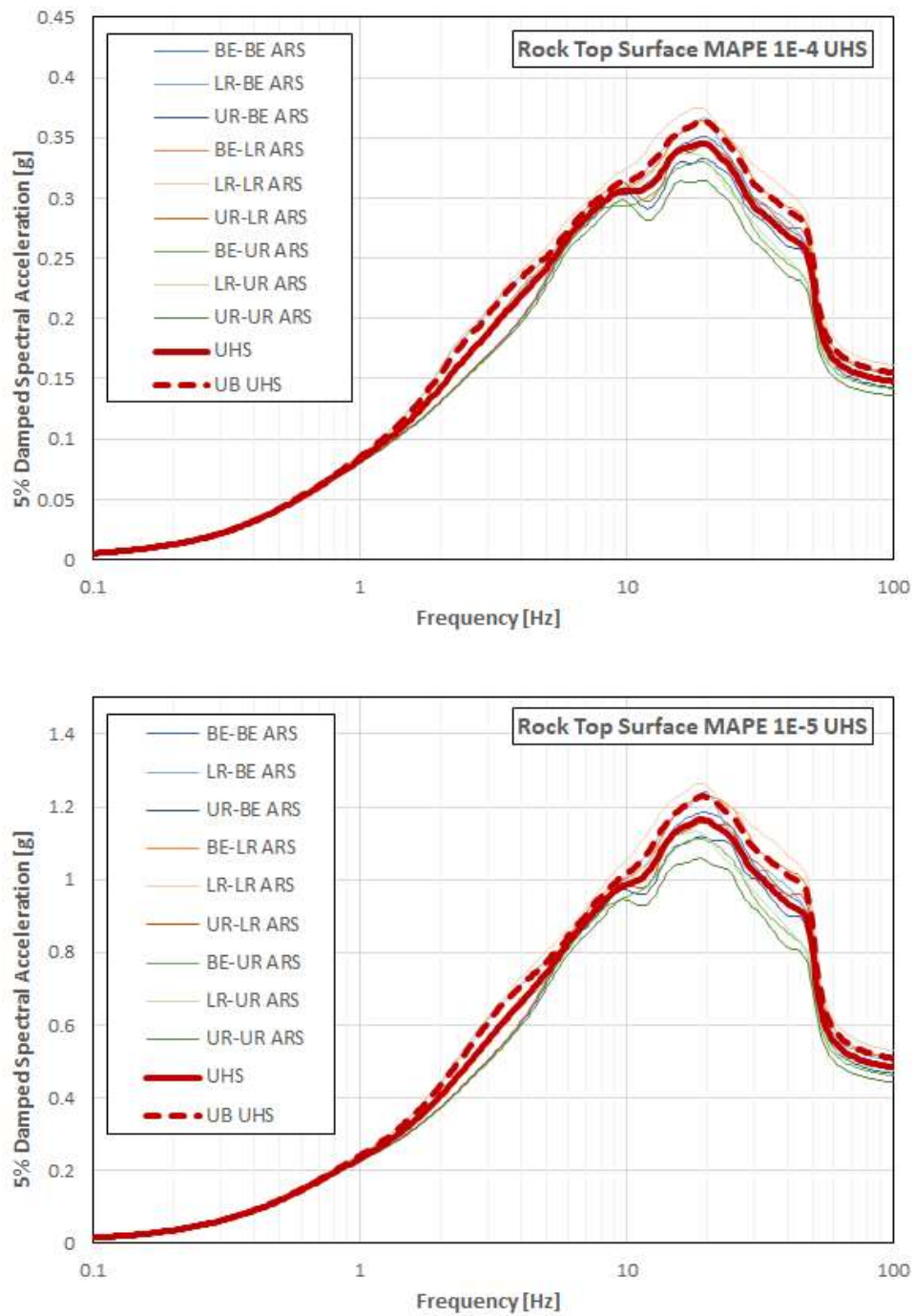


Figure 3.3-7: Rock Top Surface Uniform Hazard Response Spectra

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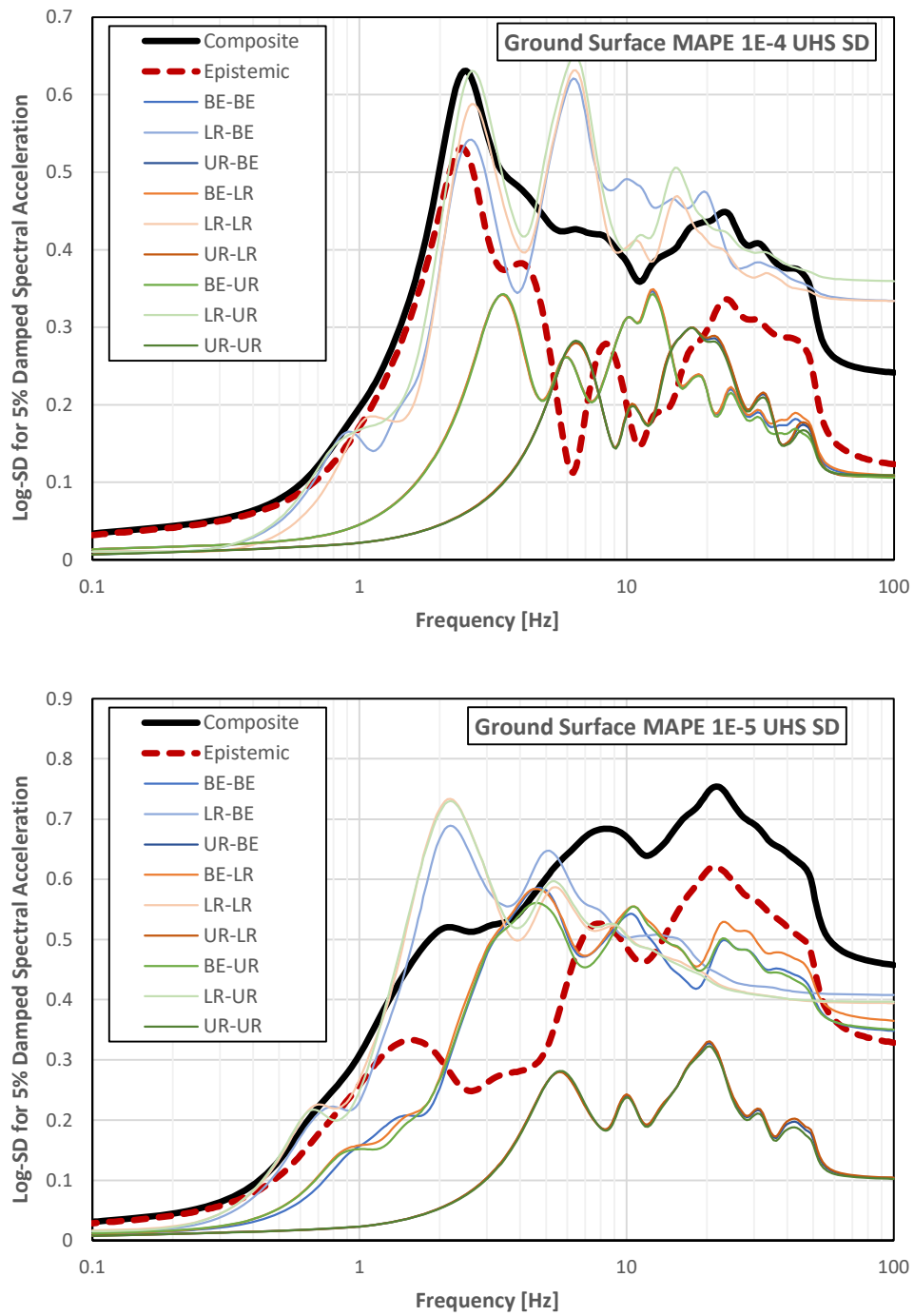
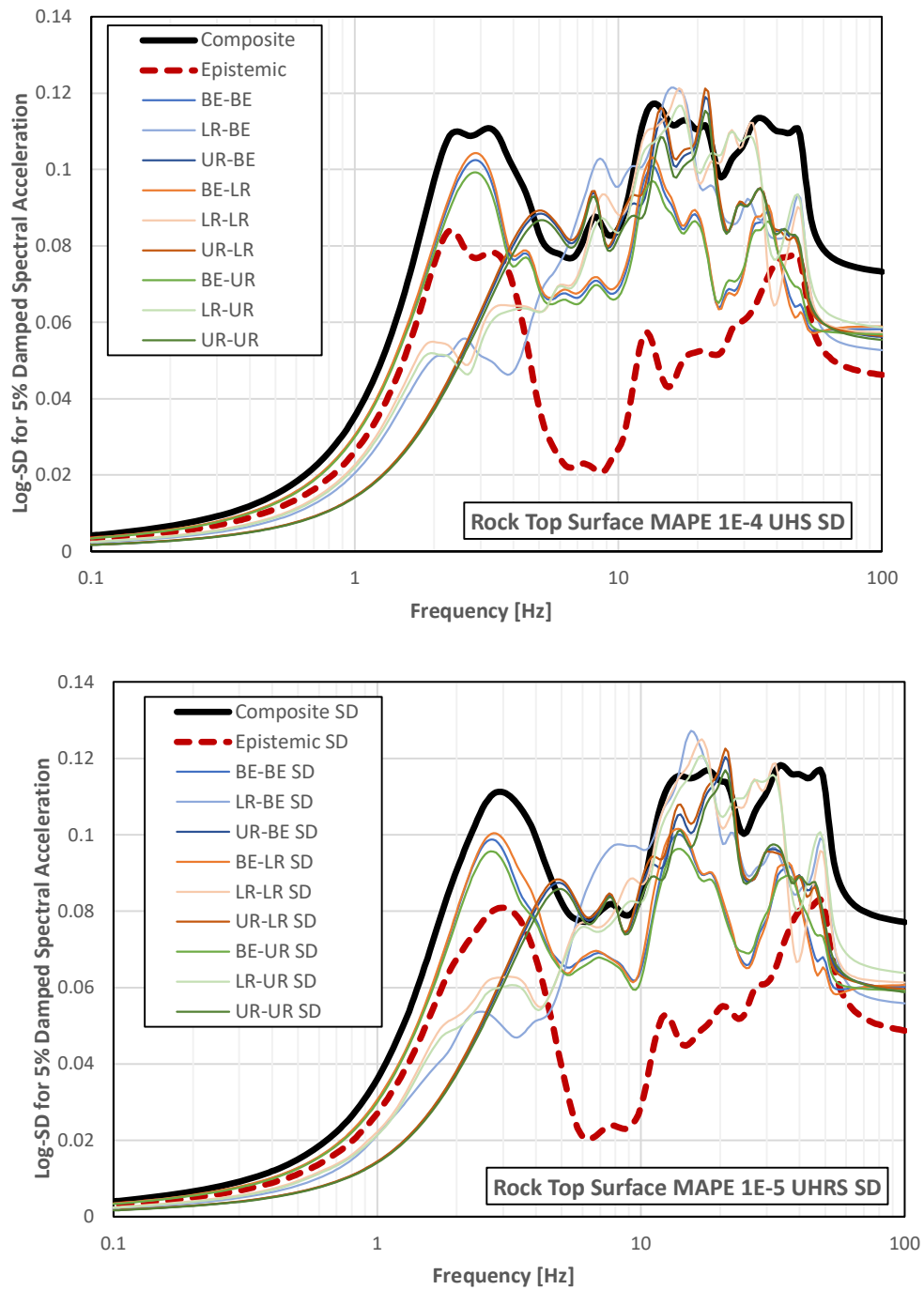


Figure 3.3-8: Ground Surface Composite and Epistemic Log-Normal Standard Deviations



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**Figure 3.3-9: Rock Top Surface Composite and Epistemic Log-Normal Standard Deviations**



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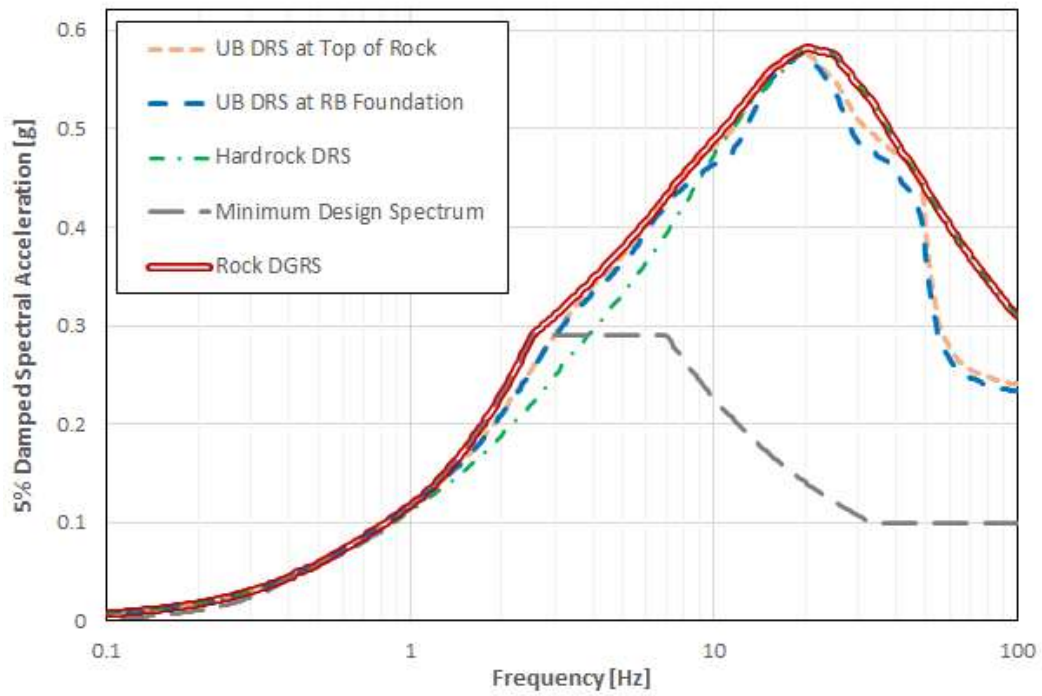
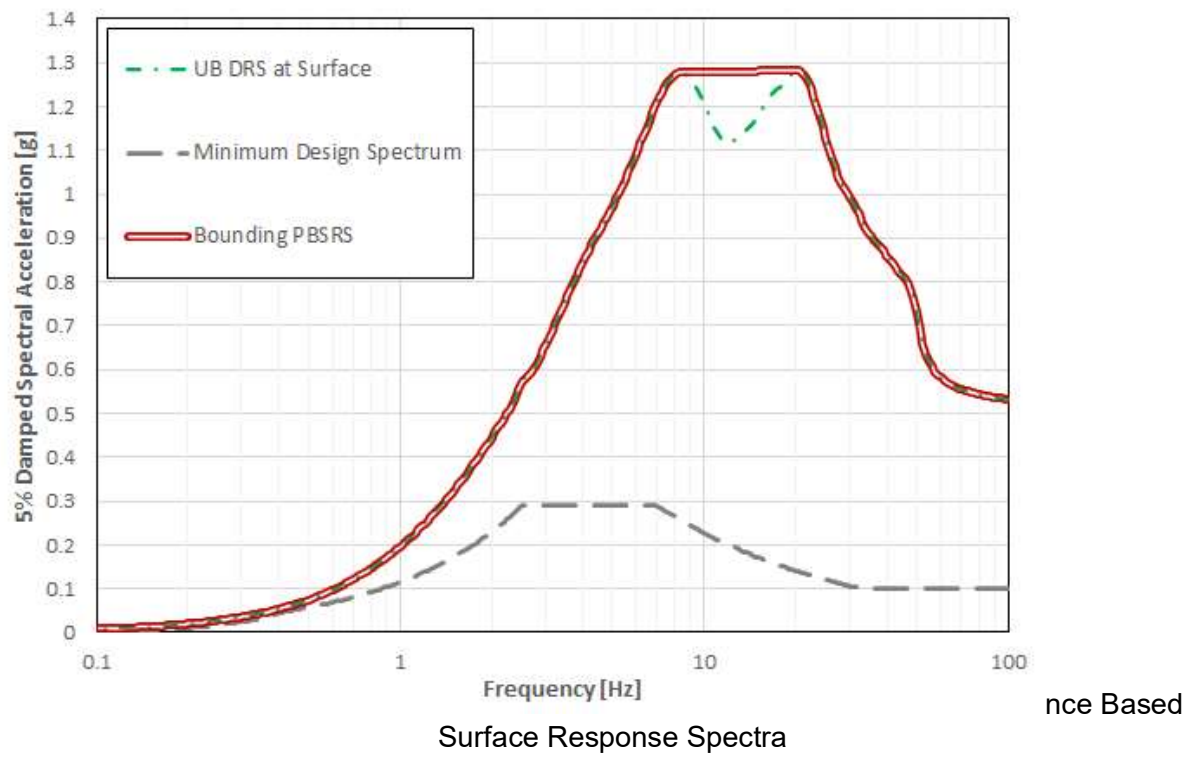
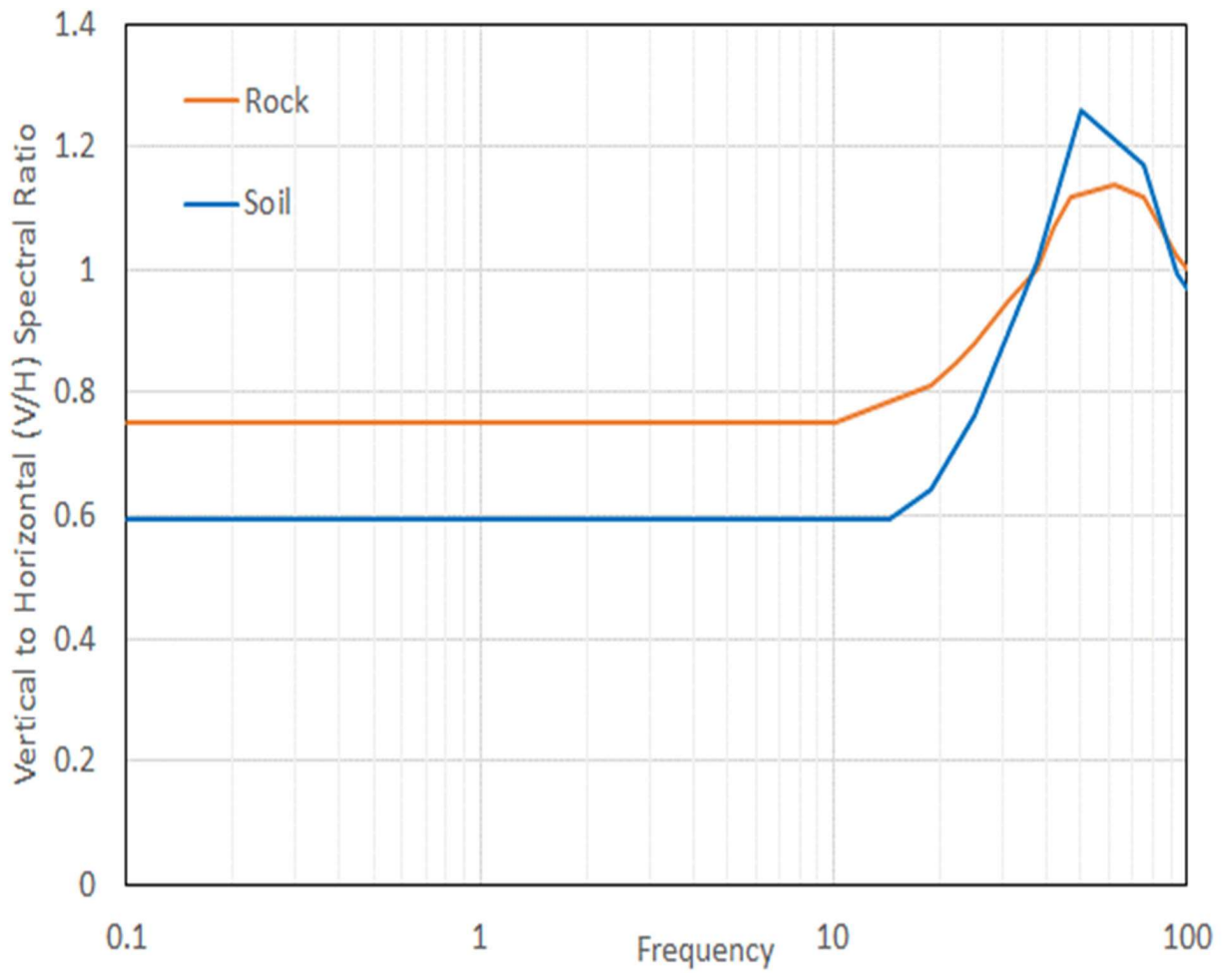


Figure 3.3-10: Bounding Horizontal Rock Ground Motion Response Spectra



**Figure 3.3-11: Bounding Horizontal Performance Based Surface Response Spectra**



**Figure 3.3-12: Vertical to Horizontal (V/H) Spectral Ratios**

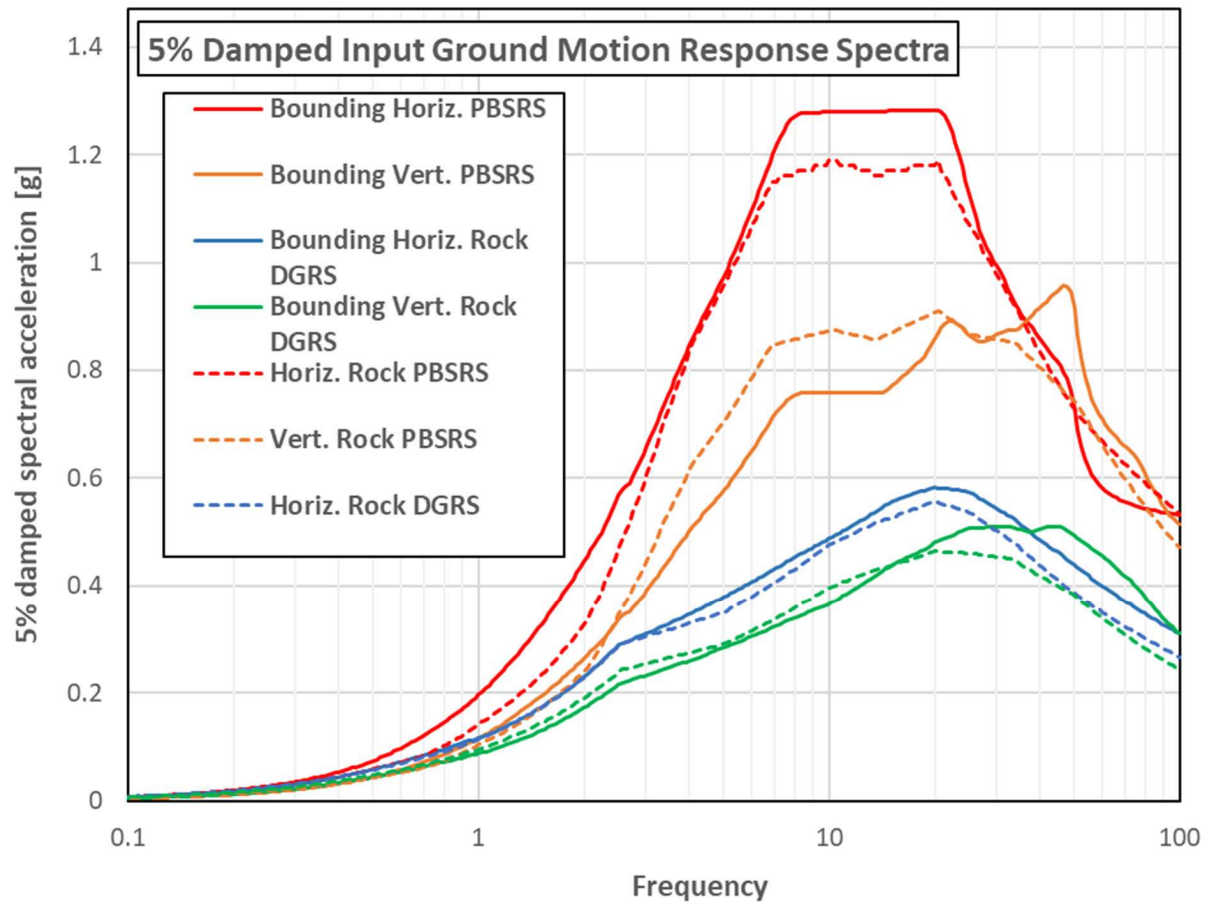
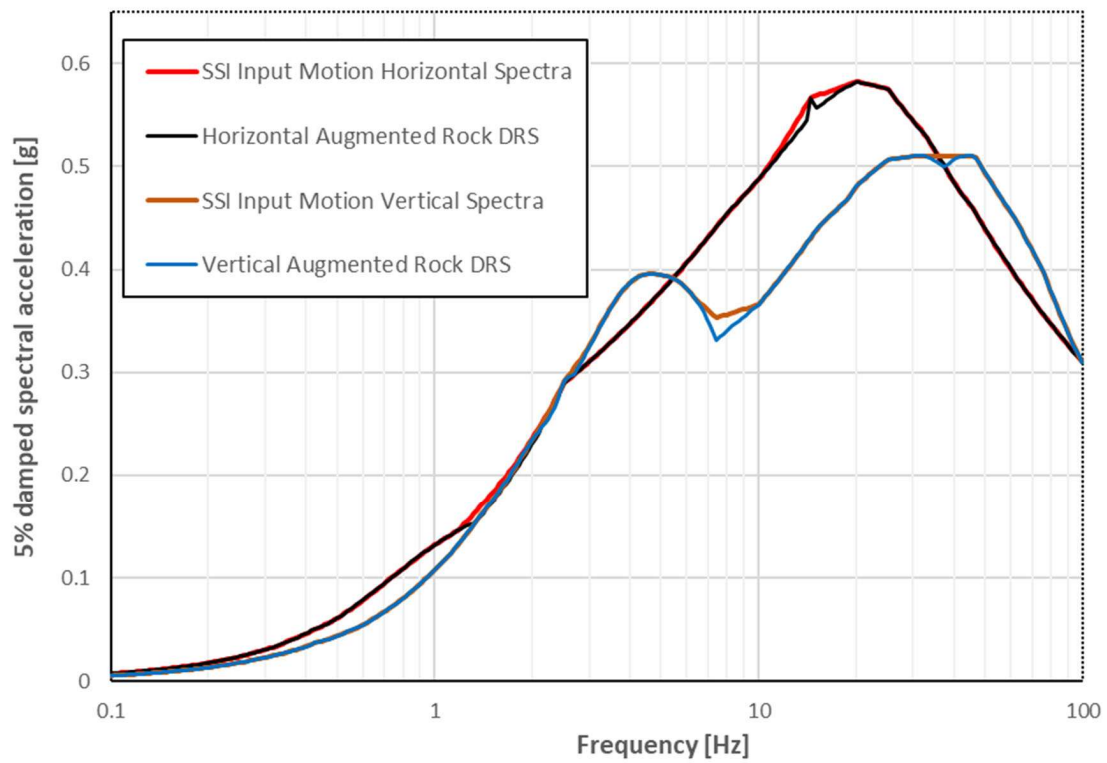
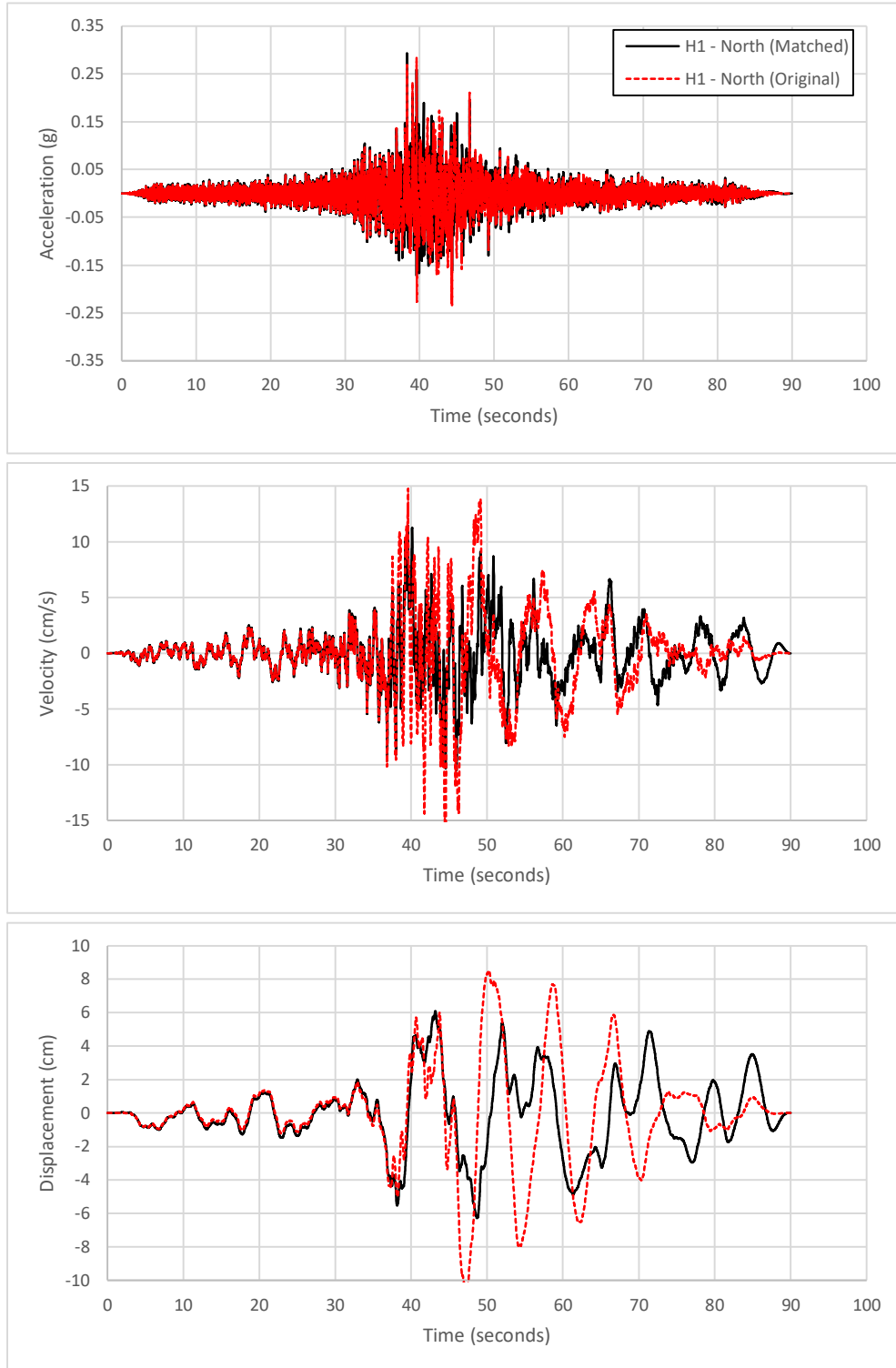


Figure 3.3-13: Comparison of Bounding to Updated Ground Motion Design Response Spectra



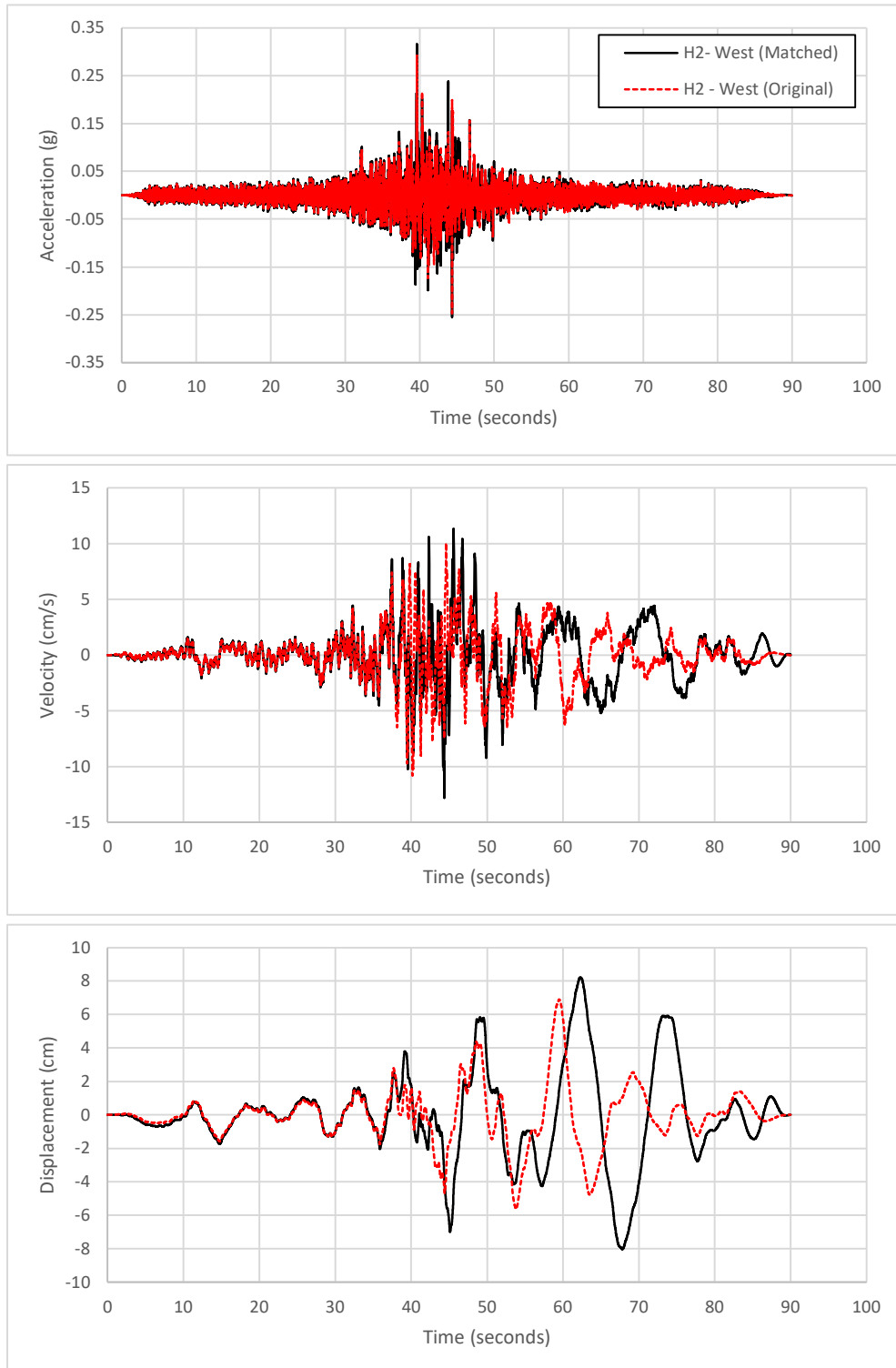
**Figure 3.3-14: Augmented and Smoothed Horizontal and Vertical Rock Design Ground Response Spectra**

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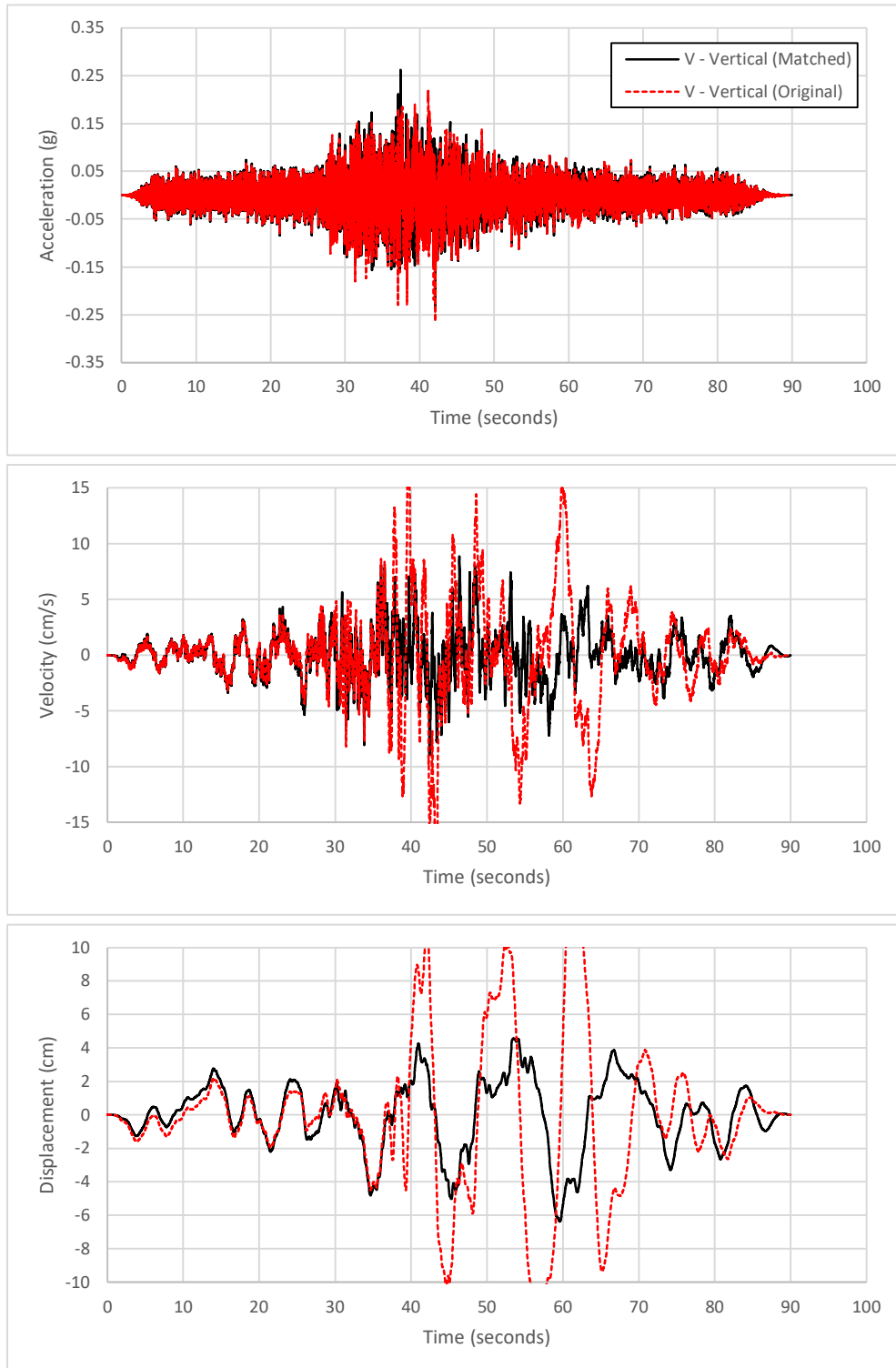
**Figure 3.3-15: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 North**

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**Figure 3.3-16: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 West**

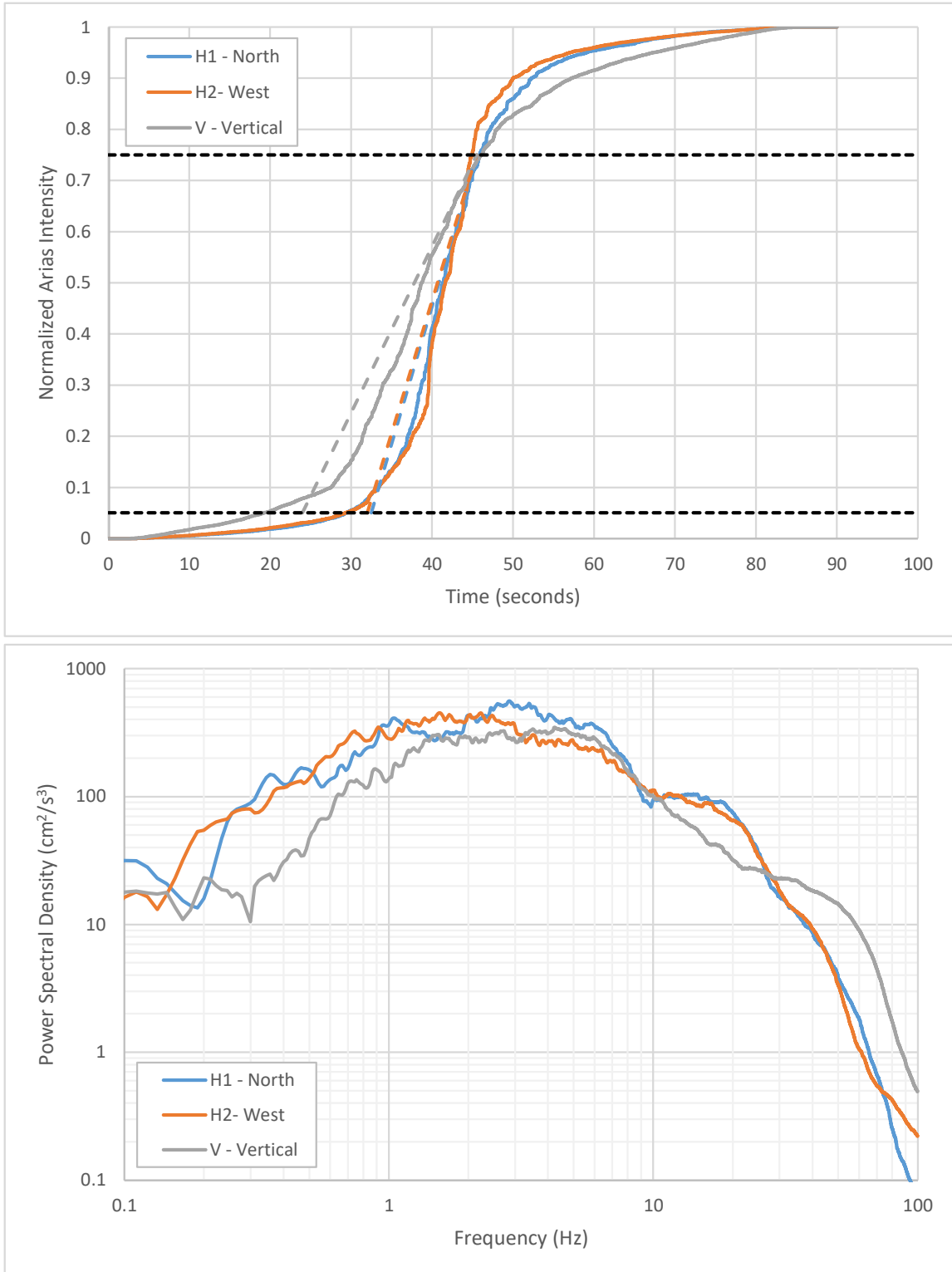
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**Figure 3.3-17: Vertical Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 Vertical**

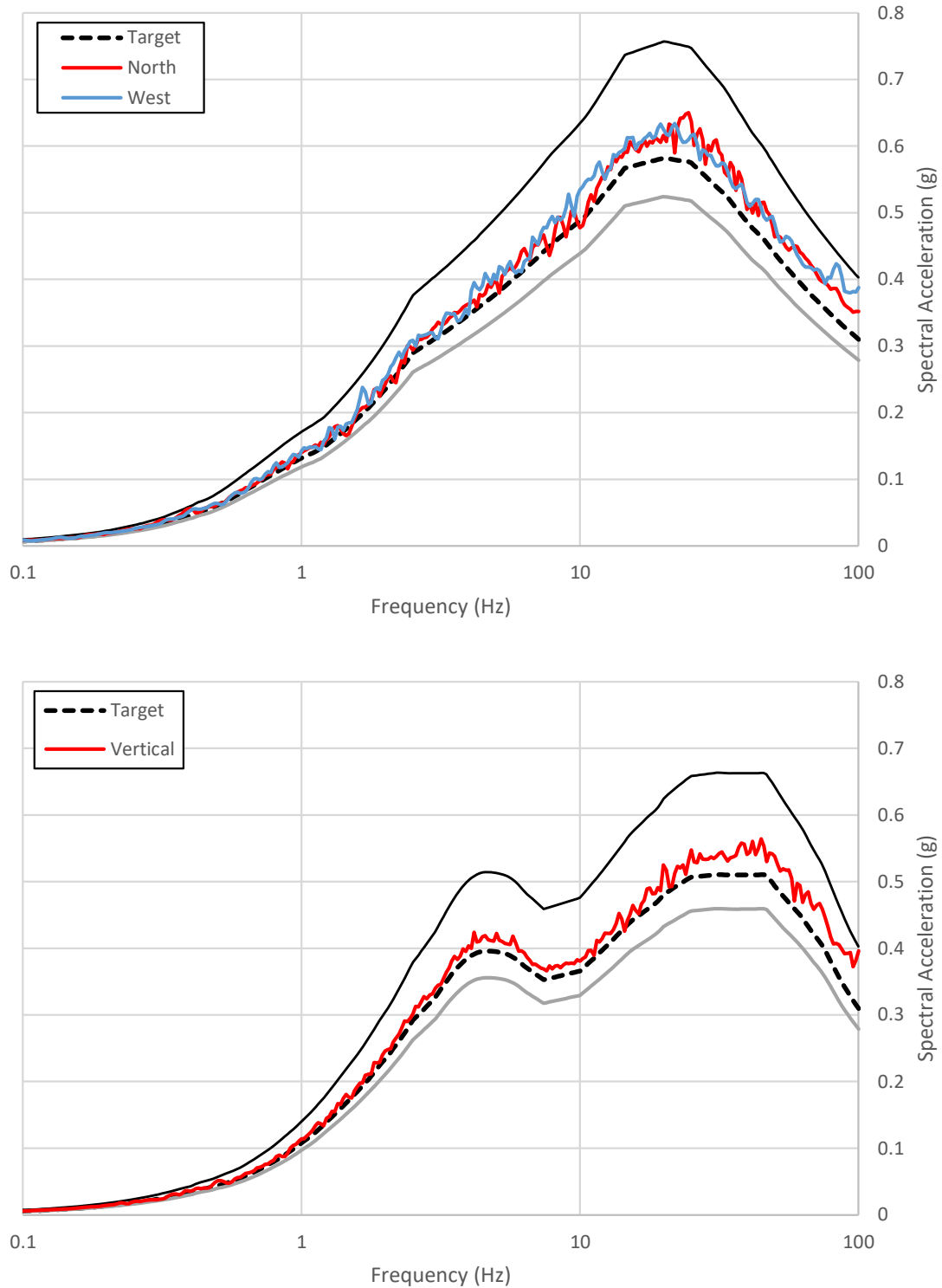


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**Figure 3.3-18: Normalized Arias Intensity and Power Spectral Density Function for Response Spectrum Matched HWA026 Acceleration Time Histories**

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**Figure 3.3-19: 5% Damped Response Spectra for Response Spectrum Matched HWA026 Acceleration Time Histories**

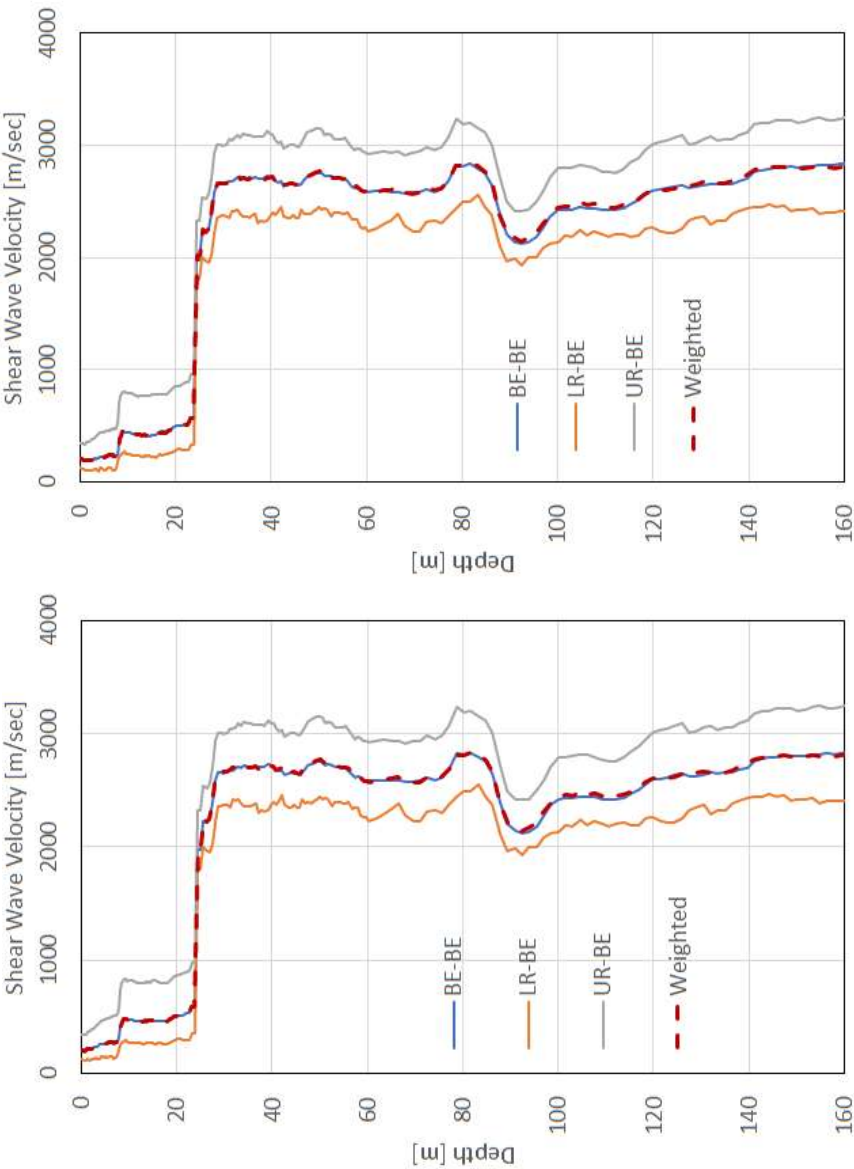
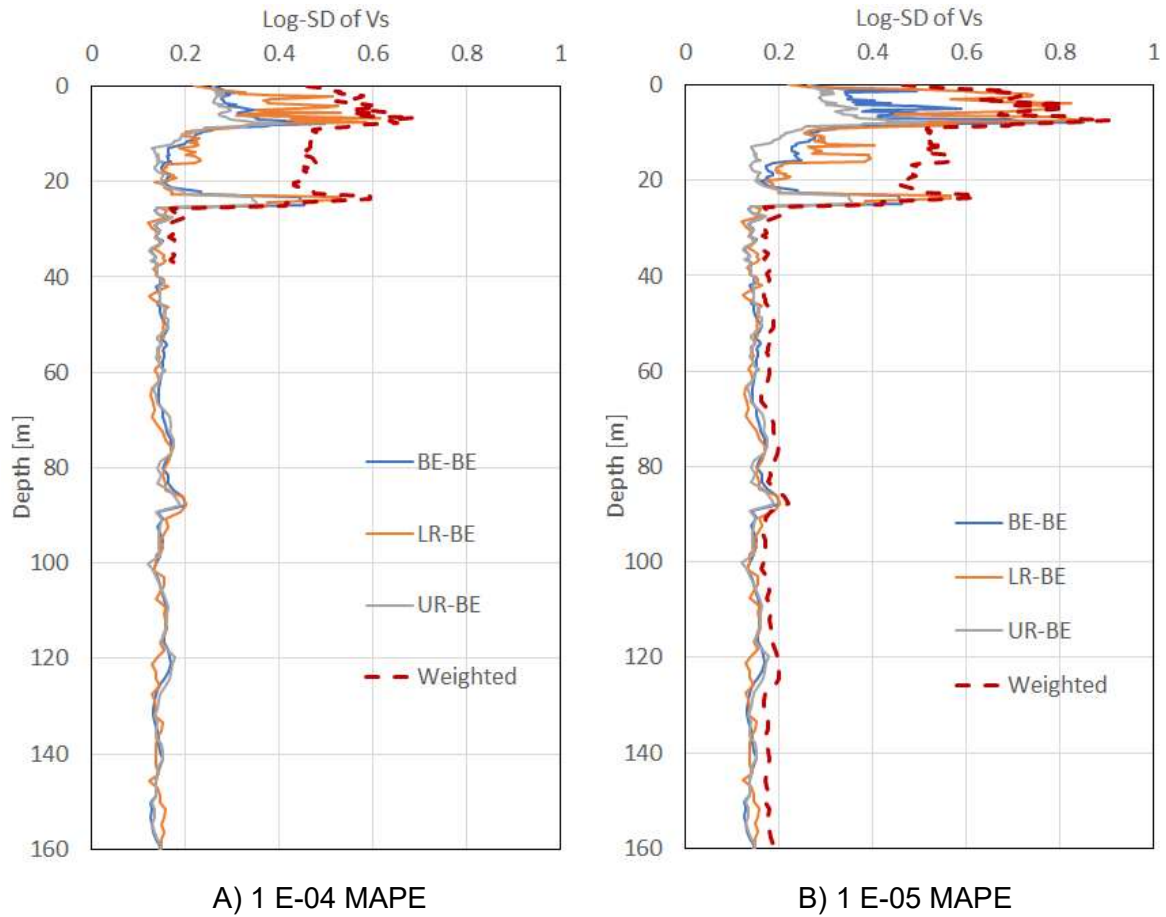


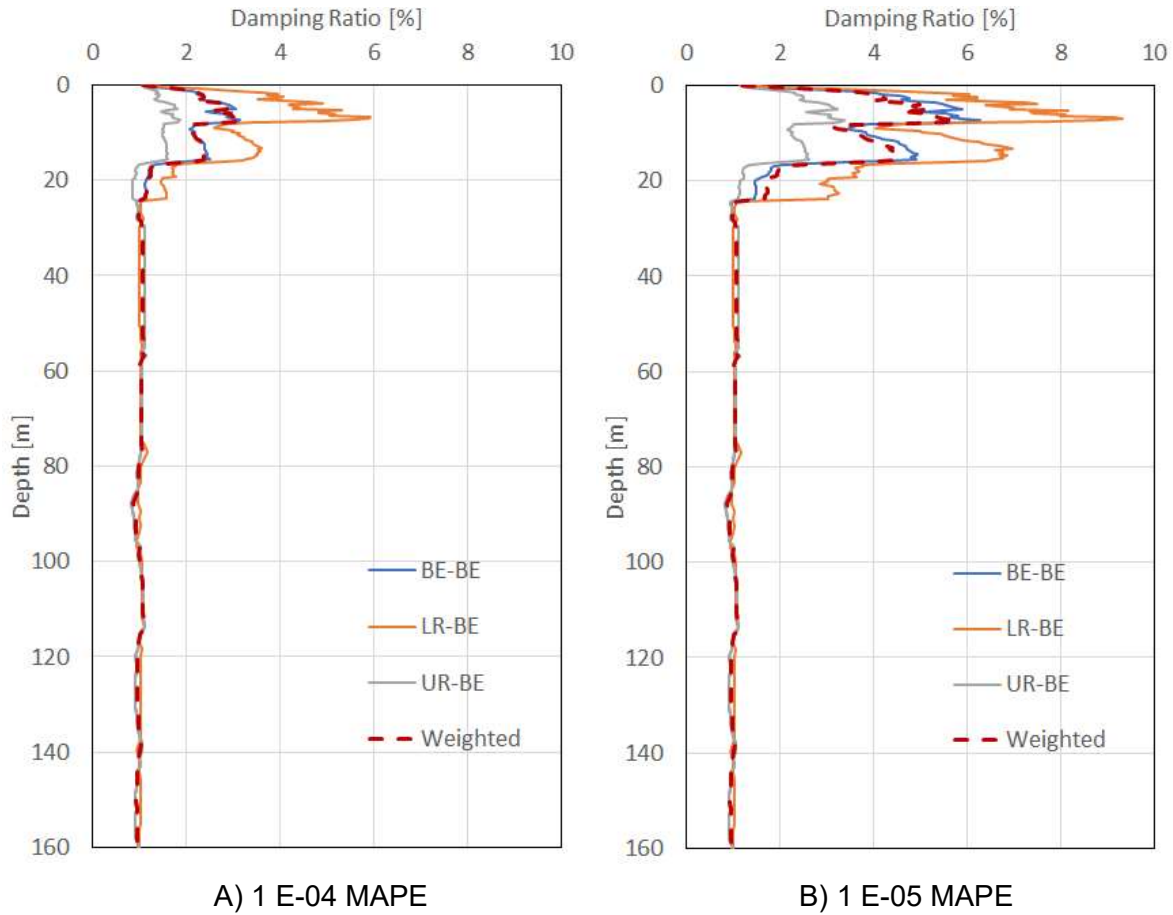
Figure 3.3-20: Logarithmic Mean of Strain-Compatible Shear Wave Velocities

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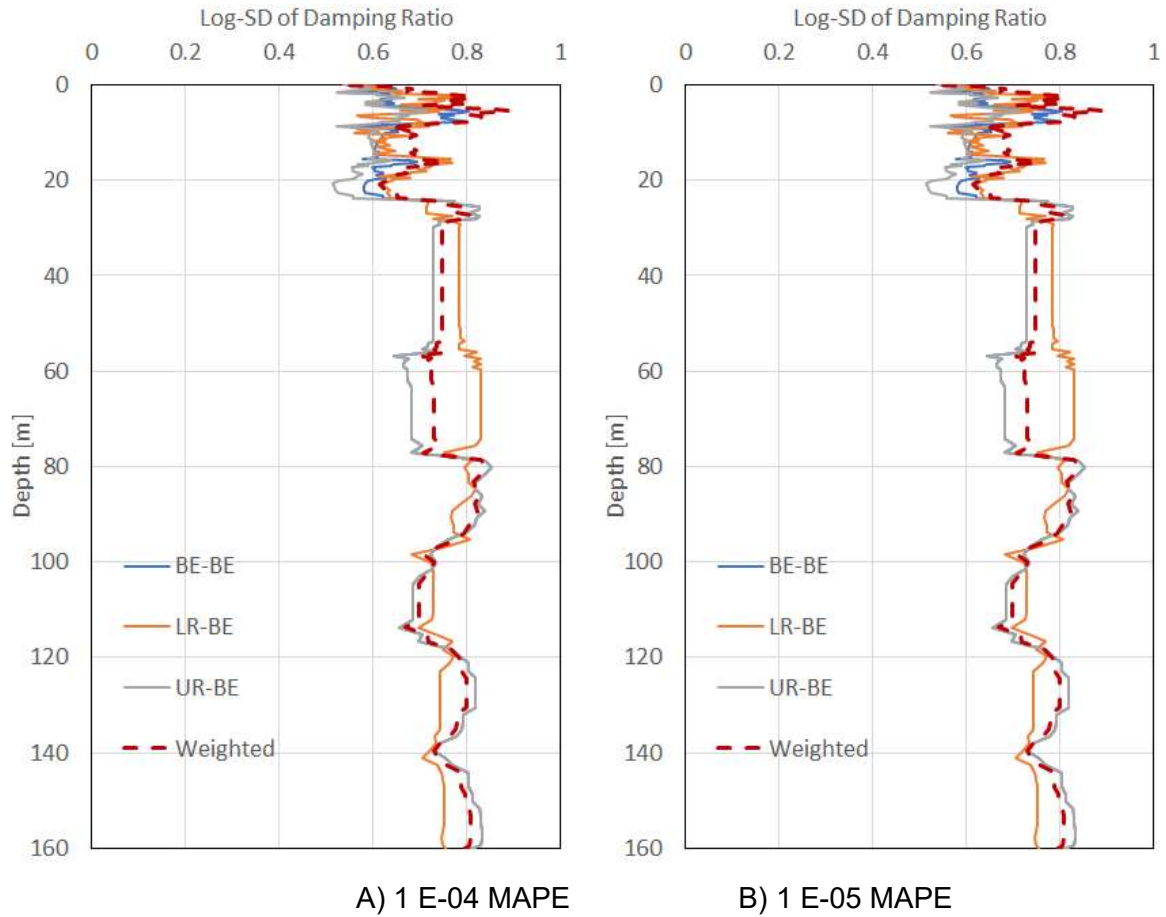
**Figure 3.3-21: Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocities**

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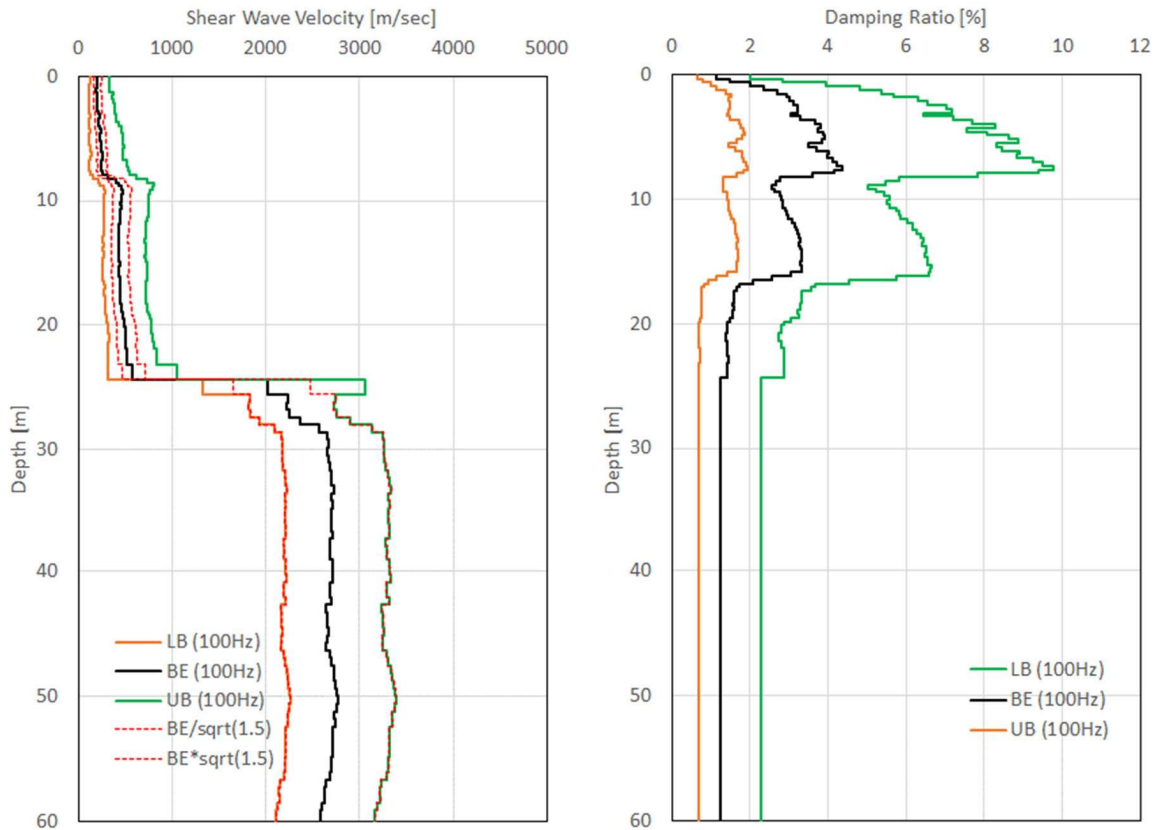
**Figure 3.3-22: Logarithmic Mean of Strain-Compatible Damping Ratios**

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**Figure 3.3-23: Logarithmic Standard Deviation of Strain-Compatible Damping Ratios**

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a) Shear Wave Velocity

b) Shear Wave Damping Ratio

**Figure 3.3-24: Strain-Compatible Shear Wave Velocity and Damping  
Using 100 Hz Interpolation**

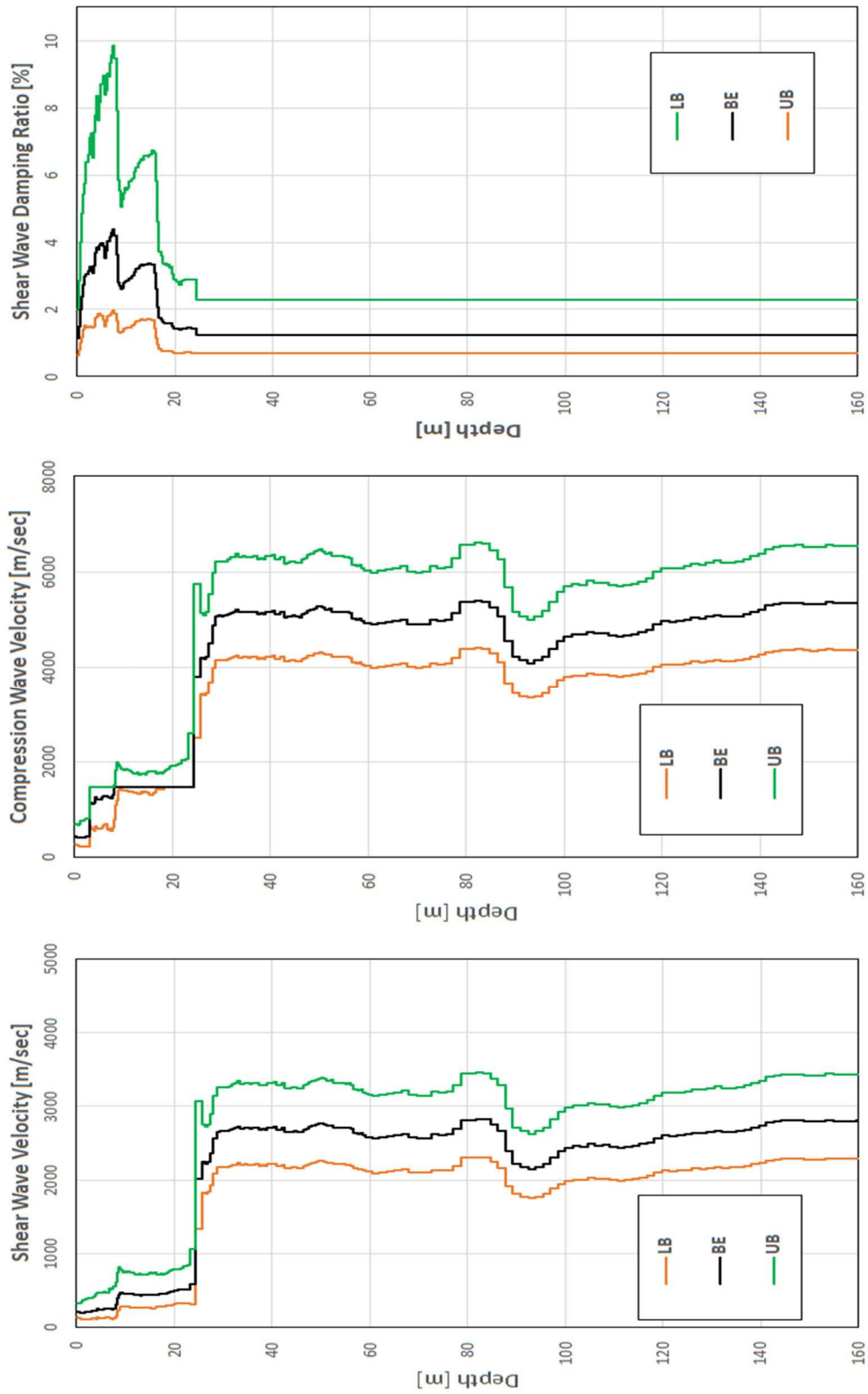


Figure 3.3-25: Subgrade Profiles for Bounding BWRX-300 Seismic Analyses



### **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. For BDBA 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:

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- 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 -

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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,

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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

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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.

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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.

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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.

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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.

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- 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

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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

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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-33914, 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.

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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

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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.

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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.

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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

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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

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- 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.

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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.



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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's.

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's 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's and evaluate the consequences of the DEC's.

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's.

A reasonable level of survivability of the structure under postulated DEC's 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's.

##### **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's.

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's.

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

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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

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- 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."

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- 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.
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- 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)

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- 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.

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- 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.



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**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

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**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

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**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	$0.67F_y$

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**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.

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**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

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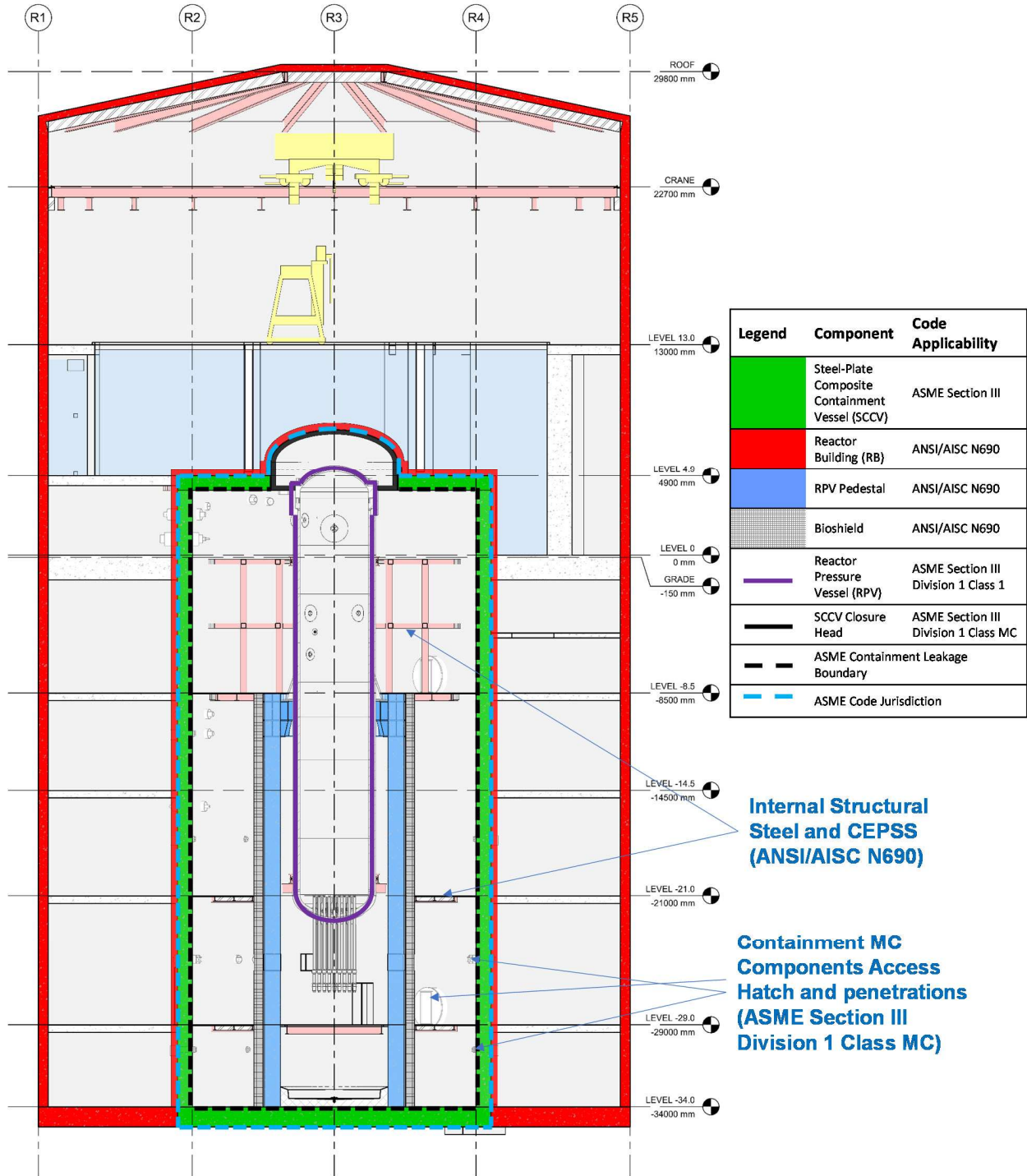
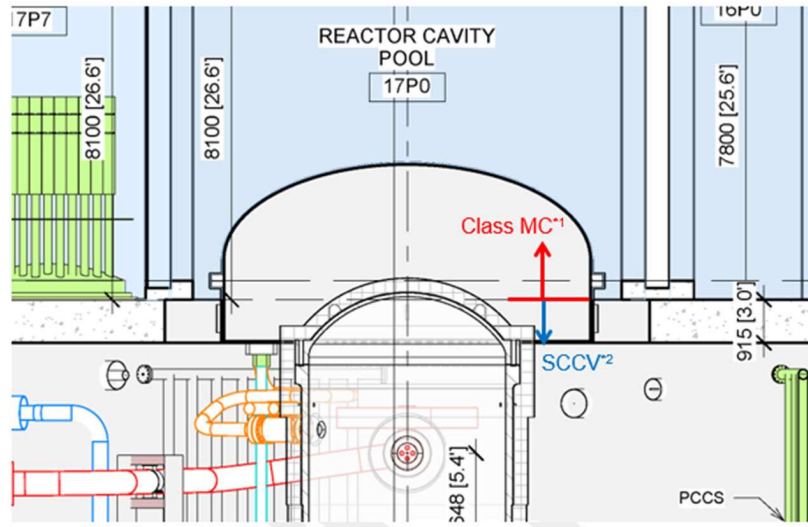


Figure 3.5-1: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

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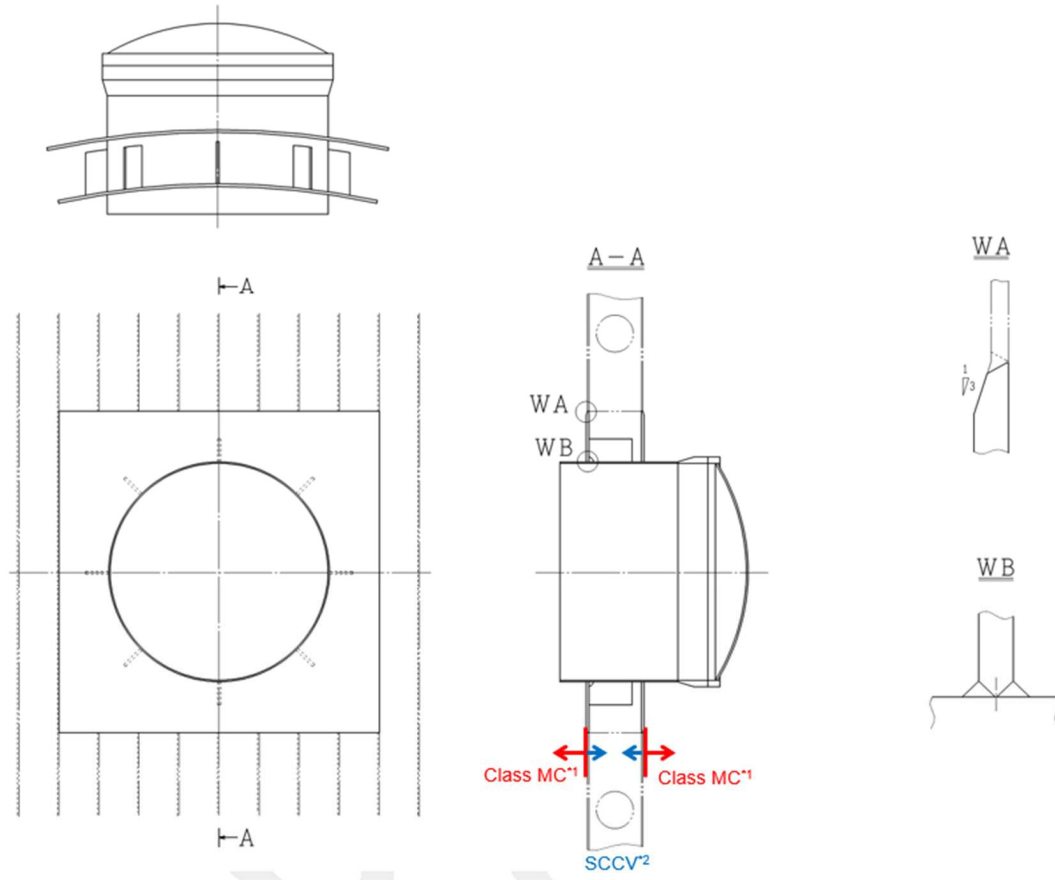
**Figure 3.5-2: Containment Closure Head Structure Boundary**

\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

\*2: Is designed in accordance with NEDC-33926P



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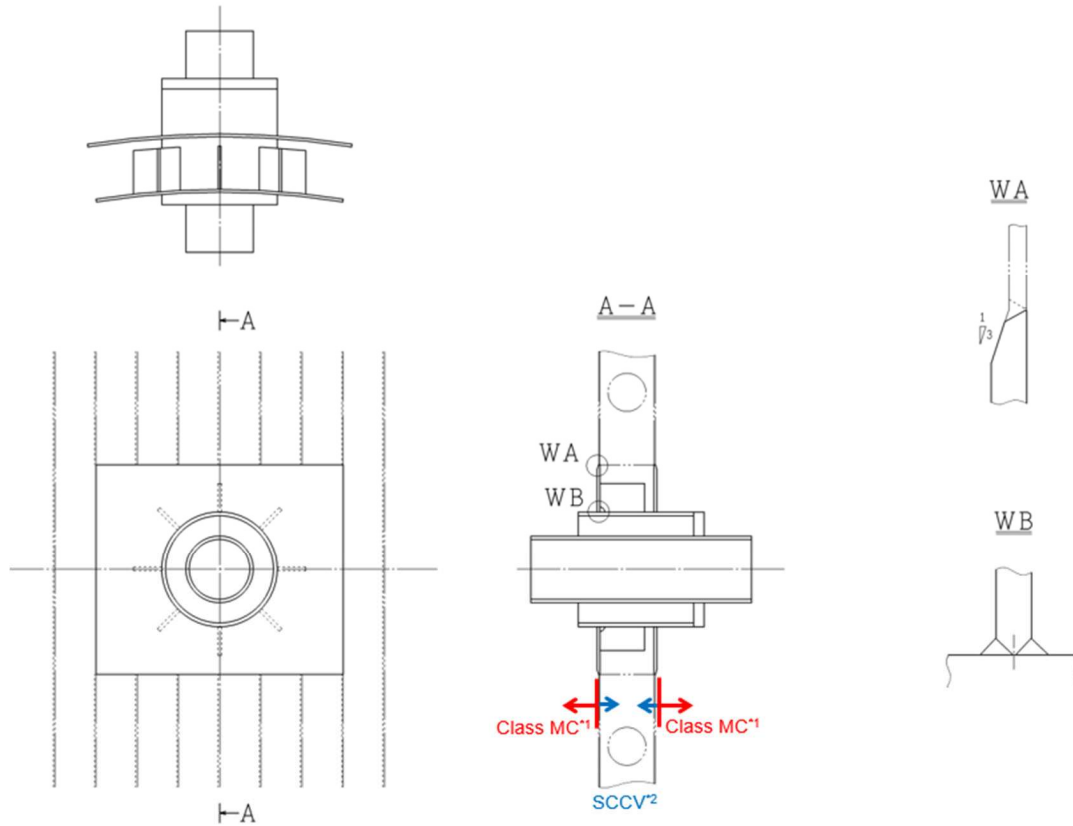


**Figure 3.5-3: Access Hatch Code Jurisdictional Boundary**

\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

\*2: Is designed in accordance with NEDC-33926P

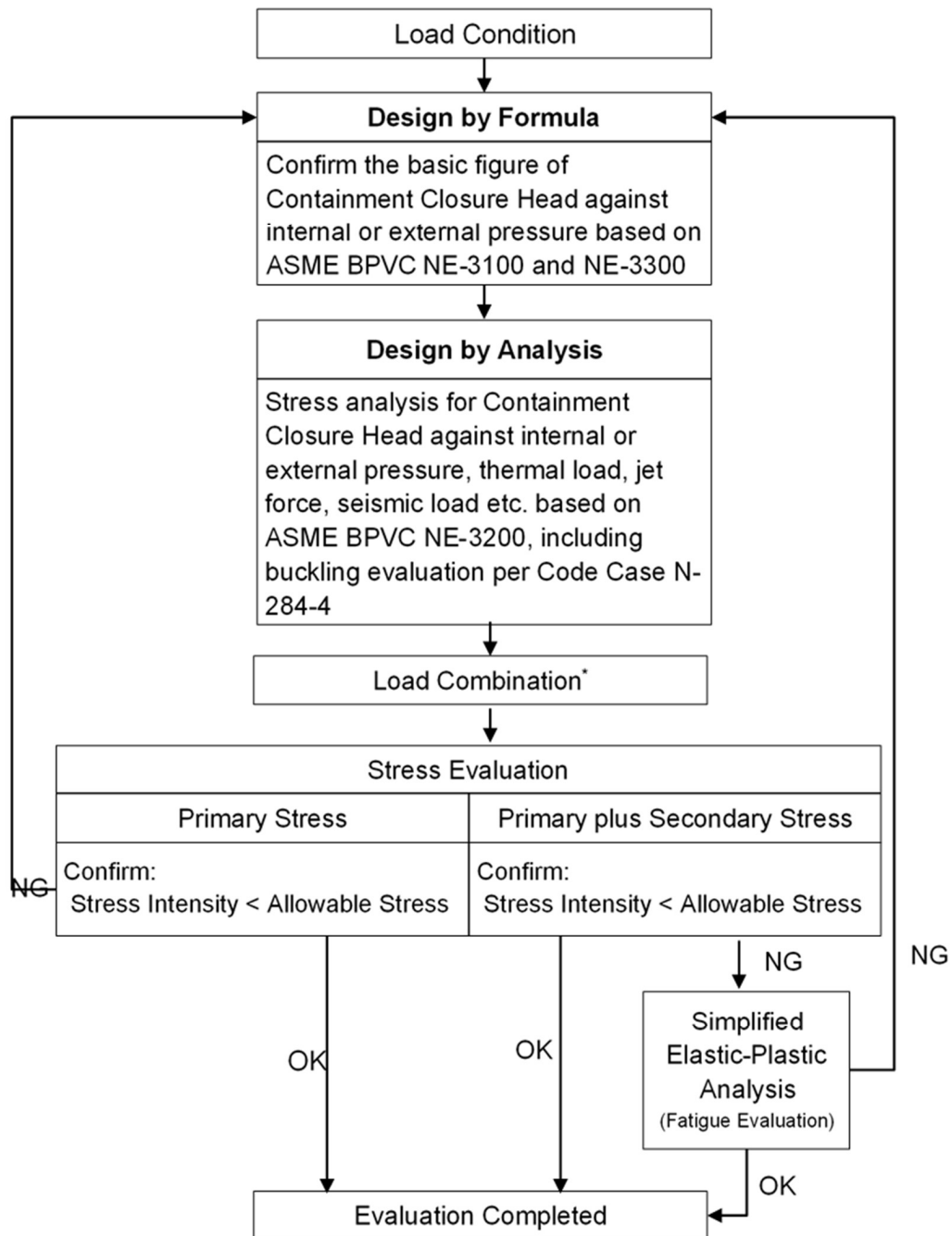
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**Figure 3.5-4: Penetrations Jurisdictional Boundary**

\*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

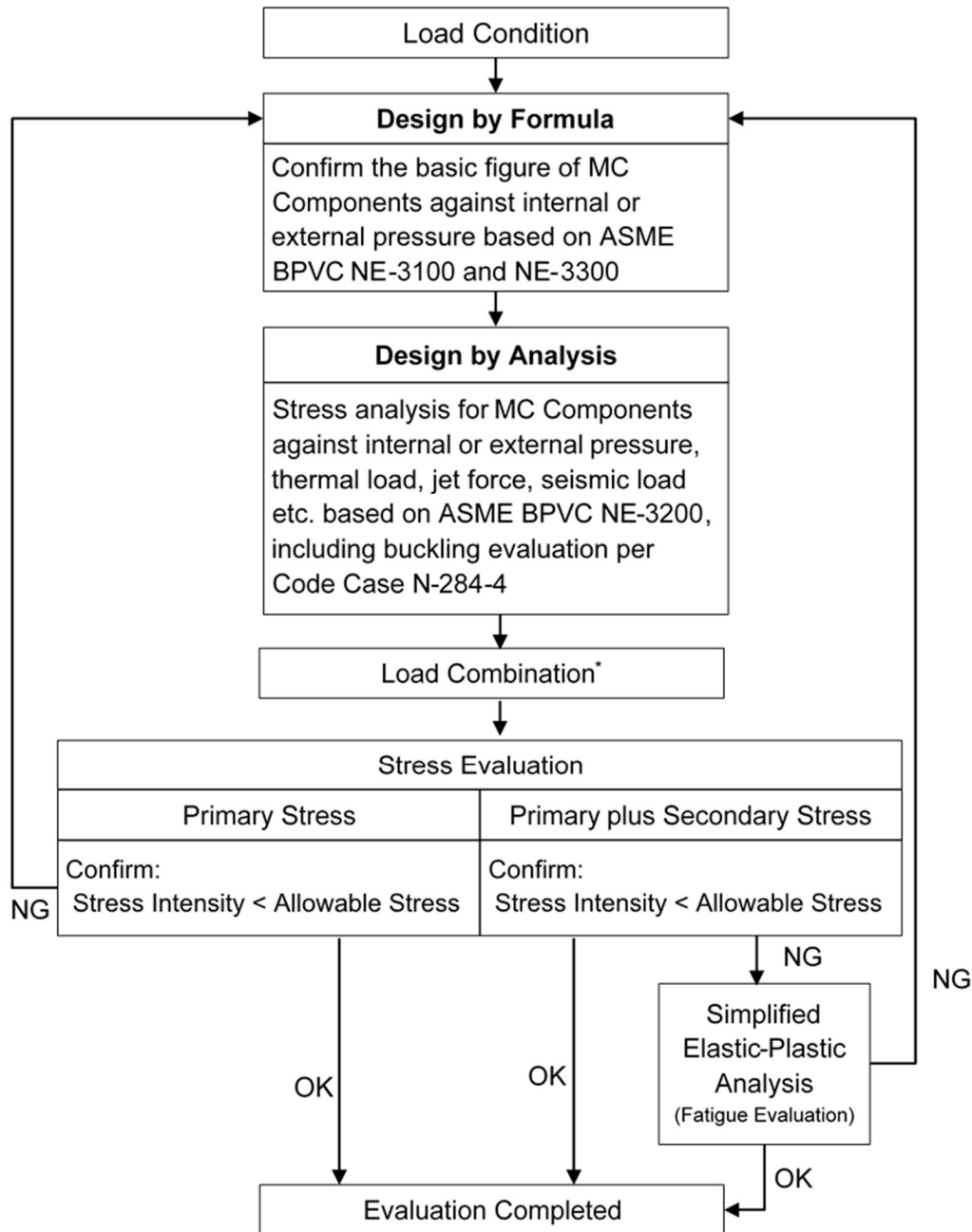
\*2: Is designed in accordance with NEDC-33926P



**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



**Figure 3.5-6: Design Procedures for the MC Components**

\*: 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)

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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 3.6.1.1, 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.

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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

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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 mainstream 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.

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**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.

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- 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.

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**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

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**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



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- 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  $(1.5 \text{ to } 1.67) * DBE$ .

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**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.	$F \leq 1\text{E-}05$

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**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

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**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

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**Table 3.6-6: Anticipated Operational Occurrences (DC-2)**

Description	Number of Cycles/60 Years
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

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**Table 3.6-7: Design Basis Accidents (DC-3)**

<b>Description</b>	<b>Number of Cycles/60 Years</b>
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)

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**Table 3.6-8: Design Extension Condition (DC-4)**

Description	Number of Cycles/60 Years
Bounding Transient without Scram	$\leq 0.001$
Pipe Rupture – Loss-of-Coolant Accident	$\leq 0.001$
Ultimate Overpressure Protection	$\leq 0.001$

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**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



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**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.

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**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.

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**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 DECAs 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

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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.

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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.

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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.

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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 envelops 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).

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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.

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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

#### **3.9.4.4.4 *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.

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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.

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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.

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- 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.

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- 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."

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**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.

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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."

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- 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.

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**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><i>Nuclear Boiler System</i></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

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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

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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

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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

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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



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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

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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

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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

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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

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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

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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

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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

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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>				



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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

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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,

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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

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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

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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.

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**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

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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  $^{239}\text{Np}$ . DECAF01A 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, DECAF01A 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

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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

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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\* Vle 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.

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**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-

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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**

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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.

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**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

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None.

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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

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- 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



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- 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).

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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).

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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.

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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

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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

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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.

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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.

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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

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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

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- 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.



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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

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- 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:

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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

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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.



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**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.

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- 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**

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September 30, 2022

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**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.

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**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

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## **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.

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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.

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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

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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.

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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-PROC-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

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- 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).

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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.

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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.

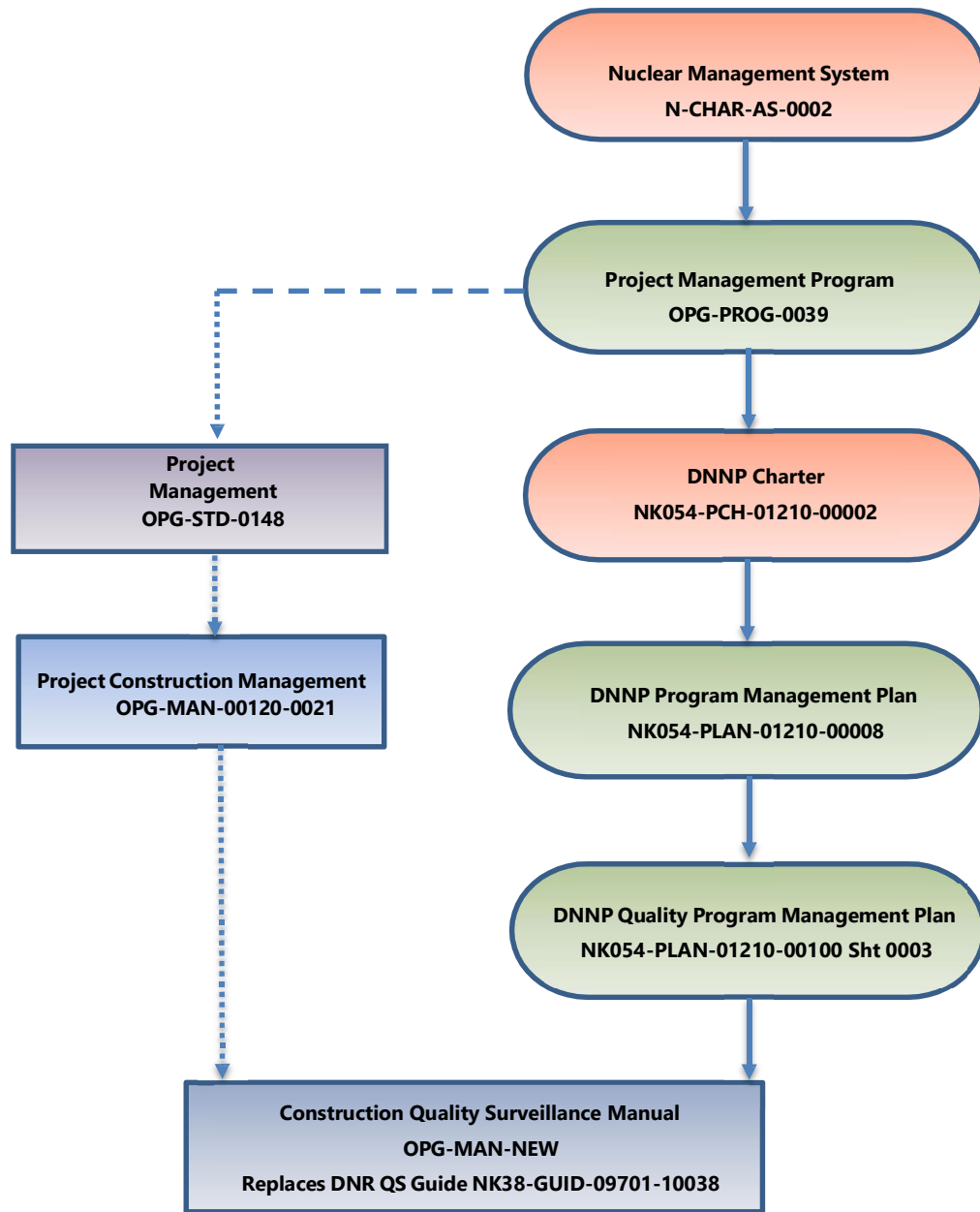


Figure 14.1-1: Quality Program Process Documents

## **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.



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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.

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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.

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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.

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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.

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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.

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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



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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.

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- 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.

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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.

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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).

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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.

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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.

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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.

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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:

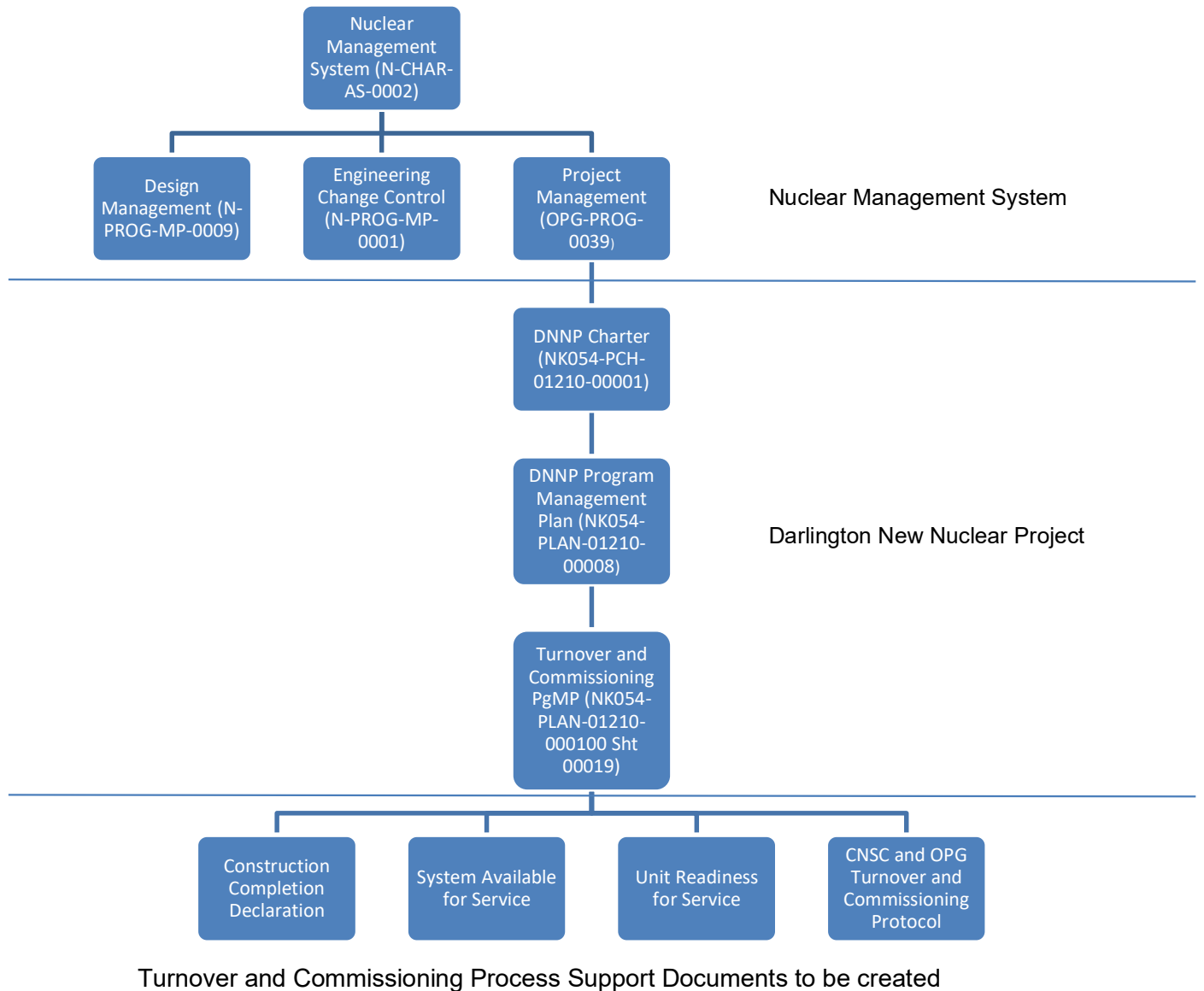
1. SSC have been constructed as per design and QA requirements are satisfied
2. SSC are tested to provide assurance that the reactor facility has been properly designed, constructed, and is ready for safe operation
3. SSC operation in accordance with the assumptions and intent of the Commissioning Program 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

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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.

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**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

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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



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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.

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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.

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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.

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**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.

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- 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."