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

REVISION SUMMARY

Revision #	Section Modified	Revision Summary
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.

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.

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.	

ACRONYM LIST

Acronym	Explanation
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

Acronym	Explanation
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

Acronym	Explanation
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

Acronym	Explanation
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

Acronym	Explanation
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Seismic Design Category
SDE	Site Design Earthquake
SEI	Structural Engineering Institute
SIL	Safety Integrity level
SIR	Seismic Interface Restraint
SIT	Structural Integrity Test
SMAMP	Structures Monitoring and Aging Management Program
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SRA	Site Response Analysis
SRSS	Square-Root-of-the Sum of the Squares
SSC	Structures, Systems, and Components
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UHRS	Uniform Hazard Response Spectrum
UL	Underwriters Laboratory
UR	Upper Realization
USNRC	U.S. Nuclear Regulatory Commission
V/H	Vertical to Horizontal
ZPA	Zero Period Acceleration

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

3.0 SAFETY OBJECTIVE AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS

This chapter introduces the safety objectives and the Safety Strategy to meet those objectives for the design and construction of the Boiling Water Reactor, 10th Design – 300 MWe (BWRX-300) Small Modular Reactor (SMR) facility at the Darlington site in Ontario, Canada.

Additionally, this chapter describes the methodology for classification of Structures, Systems, and Components (SSC), the design measures for protection against external and internal hazards, the general design aspects, and codes and standards applied to the BWRX-300 design to meet the requirements of the Nuclear Safety and Control Act (NSCA) and associated Canadian Nuclear Safety Commission (CNSC) Regulations and relevant Regulatory Documents.

3.1 General Safety Design Basis

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy. The objective of the Safety Strategy is to establish a design with a high level of safety. This is accomplished through incorporation of design requirements as set forth in CNSC REGDOC-2.5.2, (Reference 3.1-1) which to a large degree are based on the principles set forth in the International Atomic Energy Agency (IAEA) document SSR-2/1 (Reference 3.1-2).

The establishment of the BWRX-300 design basis is achieved through an iterative safety framework wherein the design is implemented to meet defined safety objectives and safety goals that are confirmed via deterministic and probabilistic safety analyses. Results of safety analyses then provide feedback into the design and the process is repeated as required until adequate design and regulatory safety margins are achieved.

3.1.1 Safety Objectives

In CNSC REGDOC-2.5.2 Section 4 (Reference 3.1-1), the CNSC endorses the safety objectives established by the IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles (Reference 3.1-3) which when followed ensure that reactor facilities are operated, and activities conducted to achieve the highest standards of safety that can be reasonably achieved. These safety objectives are described below:

General Nuclear Safety Objective: Reactor facilities are designed and operated in a manner that will protect individuals, society, and the environment from harm by establishing and maintaining effective defences against radiological hazards due to ionizing radiation. The general nuclear safety objective is supported by the following three complementary safety objectives:

1. Radiation Protection Objective: Radiation exposures within the reactor facility during normal operations, during anticipated operational occurrences or due to any planned release of radioactive material from the reactor facility are kept below prescribed limits and As Low As Reasonably Achievable (ALARA). Provisions are made for the mitigation of the radiological consequences of accidents.
2. Technical Safety Objective: All reasonably practicable measures are taken to prevent accidents in the reactor facility and to mitigate the consequences of events should they occur.
3. Environmental Protection Objective: All reasonably practicable mitigation measures to protect the environment during the operation of a reactor facility and to mitigate the consequences of an accident are provided. The design includes provisions to control, treat and monitor releases to the environment and minimize the generation of radioactive and hazardous wastes.

The high-level safety objectives inform the principal safety objectives in the design and safety analyses.

3.1.2 Radiation Protection and Radiological Acceptance Criteria

3.1.2.1 Radiation Protection

The BWRX-300 is designed to meet the Radiation Protection Objective by ensuring that potential radiation dose to workers and the public is kept below prescribed regulatory limits per the Radiation Protection Regulations (Reference 3.1-4) and ALARA.

This is achieved by a comprehensive and appropriately conservative source term derivation identifying radiation sources during the design phase to ensure means are provided to reduce occupational exposure during plant operation, maintenance, and decommissioning.

Safety features and measures include:

- Passive engineered safety features
- Active engineered safety features
- Administrative safety measures

Engineered safety features include shielding, containment, ventilation, remote handling, and interlocks. Administrative safety measures that reduce exposure to the hazard during planned operations include restrictions on occupancy, monitoring arrangements, pre-planning of exposure and the use of barriers and notices. Passive engineered safety measures (e.g., containment or shielding) are preferred before active engineered safety features and administrative safety measures. Human factors considerations are incorporated into the engineered and administrative measures (See Chapter 18 for details).

System design evaluations are performed in parallel with other activities to ensure systems support operational objectives. These evaluations include the development of reasonable and practical measures to achieve minimal dose to workers and the public.

Details on how radiation protection is considered in the design for operational states and accident conditions are provided in Chapter 12.

3.1.2.2 Radiological Acceptance Criteria

Limits on radiation dose are established by the CNSC through the Radiation Protection Regulations (Reference 3.1-4). The expectation established is that during normal operation, including maintenance and decommissioning, dose to workers and the public are ALARA.

Per CNSC Radiation Protection Regulations (Reference 3.1-4), the effective dose limit for a nuclear energy worker is an average of 20 mSv effective dose per year over a five-year period (100 mSv over five consecutive years), with no single year exceeding 50 mSv effective dose. The effective dose limit for a member of the public is 1 mSv per year from all sources of radiation other than natural background and medical exposures. Additional details are provided in Reference 3.1-4.

In addition to design features, administrative measures such as radiation protection and environmental protection programs are established to ensure worker and public dose is maintained below limits. Action levels are established for effluent releases and expressed in a form that compliance can be demonstrated in a practical manner. These action levels are not limiting but, are values at which actions must occur to reduce the effluent releases from the plant. Chapter 20 discusses Effluent Dose Levels to the General Public.

Deterministic safety analyses are conducted in accordance with CNSC REGDOC-2.4.1 (Reference 3.1-5) to confirm that the BWRX-300 is designed to ensure that potential radiation doses to the public from Abnormal Operating Occurrences (AOOs) and Design Basis Accidents (DBAs) (defined in Subsection 3.1.3) do not exceed dose acceptance criteria per Section 4.2.1 of CNSC REGDOC-2.5.2 (Reference 3.1-1). In the deterministic safety analysis, the committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated for a period of 30 days after the analyzed event to confirm that for AOOs and DBAs, doses are less than or equal to the following:

- 0.5 millisievert (mSv) for any AOO or
- 20 mSv for any DBA

Chapter 15, Subsection 15.3.1, describes the dose calculation methodology used in the deterministic safety analysis. Results of the analyses are summarized in Section 15.7 demonstrating that the radiological consequences of the analyzed events do not exceed the acceptance criteria for AOOs and for DBAs.

3.1.2.3 Safety Goals

In addition to the deterministic dose acceptance criteria, Probabilistic Safety Analysis (PSA) is used to assess risks posed by reactor facility operation through the application of quantitative safety goals. These include core damage frequency, and small and large release frequency.

Core damage frequency is a measure of the capability of the design to prevent an accident that leads to core damage. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of reactor facilities. The quantitative goals as established by CNSC REGDOC-2.5.2, Section 4.2.2 (Reference 3.1-1) are:

- **Core damage frequency** - The sum of frequencies of all fault sequences that can lead to significant core degradation shall be less than 1E-5 per reactor-year.
- **Small release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E15 becquerels of Iodine-131, shall be less than 1E-5 per reactor-year.
- **Large release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E14 becquerels of Cesium-137 shall be less than 1E-6 per reactor-year.

The PSA is described in detail in Chapter 15, Section 15.6, Probabilistic Safety Analyses.

3.1.3 Plant States Considered in the Design Basis

The range of conditions and events considered are categorized into plant states based on their frequency of occurrence. Plant states include operational states and accident conditions. Operational states included in the design basis are Normal Operation and AOOs. Accident conditions considered in the design basis are DBAs. Design Extension Conditions (DECs) are accident conditions considered in the design but are outside of the design basis based on their lower expected frequency of occurrence.

These four plant states considered in the BWRX-300 Safety Strategy as described below are consistent with CNSC REGDOC-2.5.2, Section 7.3 (Reference 3.1-1):

- **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.)

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 DECs. This is accomplished through the operating procedures described in Chapter 13 and through complementary design features described in Chapter 15, Appendix 15B.

The general approach to defining the design basis for the BWRX-300 involves establishing the plant states described above, identifying the PIEs leading to a deviation from normal operation and categorizing mitigating functions based on their ability to prevent and mitigate the progression of events ensuring that the safety objectives are met.

3.1.4 Prevention and Mitigation of Accidents

The design of the BWRX-300 includes provisions to prevent and to mitigate the consequences of accidents and to ensure that the likelihood that an accident will have harmful consequences is extremely low.

The primary means of preventing and mitigating the consequences of accidents is through the application of Defence-in-Depth (D-in-D). The application of D-in-D for the BWRX-300 design is described below in Subsection 3.1.6.

3.1.5 Fundamental Safety Functions

The design of the BWRX-300 fulfills Fundamental Safety Functions (FSFs) at all plant states (defined in Section 3.1.3) which ensures the design meets the safety objectives consistent with CNSC REGDOC-2.5.2, Section 6.2 (Reference 3.1-1). The FSFs for the BWRX-300 are:

- Control of reactivity
- Removal of heat from the fuel (in the reactor, during fuel storage and handling, and including long-term heat removal)
- Confinement of radioactive materials, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental releases

The FSF prevent or mitigate radiological releases by ensuring the physical barriers to releases (fuel matrix, fuel cladding, Reactor Coolant Pressure Boundary (RCPB), and containment) remain effective. In addition to the protection of barriers, a means of monitoring the status of key plant parameters is provided for ensuring that the FSF are fulfilled. From this perspective, the monitoring function is treated as inherent to the design of the FSF. Other considerations for the monitoring function are as follows:

1. If a manual operator action plays a role in performing an FSF, the monitoring function of the equipment used to display key plant parameters that are necessary for the operator to perform the manual action successfully are also considered part of the FSF.
2. Certain monitoring functions allow the operator to confirm ongoing effectiveness of the FSFs during all plant states, to implement post-accident procedures, and to make decisions in support of emergency planning.
3. Post-Accident Monitoring (PAM) is important for operator decision making such as taking manual actions and implementing functions. Therefore, the designation, treatment and display of certain plant parameters or measurements as post-accident monitoring variables is a supporting design feature.
4. A minimum set of monitoring functions and display of parameters that do not support the operator actions are provided to support accident assessment.

Preservation of the FSFs is intrinsic to BWRX-300 Safety Strategy. A systematic approach is taken to identify the FSFs and those SSC necessary to fulfill the FSFs following a PIE. This systematic approach is detailed in the D-in-D discussion below.

3.1.6 Defence-in-Depth

3.1.6.1 BWRX-300 Defence-in-Depth Concept

The implementation of D-in-D in the BWRX-300 design is the basis for the Safety Strategy for ensuring an adequate level of safety is achieved by the design.

The concept of D-in-D (consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1)) involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of a nuclear power plant, the undesirable outcome is the exposure of workers, the public or the environment to radioactivity exceeding levels determined to be safe.

There are two types of defensive layering considered:

1. Physical barriers in place to prevent the release of radioactivity: The fuel matrix, fuel cladding, RCPB, and containment. The integrity of one or more physical barriers must be maintained to prevent unacceptable releases.
2. A combination of active, passive, and inherent safety features used to minimize challenges to the physical barriers, to maintain the integrity of the barriers and, in case a barrier is breached, to ensure the integrity of the remaining barriers.

While the physical barriers themselves represent multiple layers of defence against radioactive releases, in the BWRX-300 D-in-D application, the physical barriers are not themselves referred to as “defense lines”. That term is reserved for the layers of defence comprising features, functions and practices that protect the integrity of the barriers. The D-in-D concept applied is largely focused on identifying and organizing features, functions, and practices into defense lines without explicit acknowledgment of the physical barriers. The fundamental purpose of the defense lines is to ensure the integrity of the physical barriers by applying multiple levels of protection.

The BWRX-300 D-in-D concept uses the FSFs described above to define the interface between the defense lines and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers remain effective.

3.1.6.2 Defense lines

Five Defense Lines (DLs) (or levels), DL1 through DL5, are adopted consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1) and IAEA SSR-2/1 (Reference 3.1-2). Figure 3.1-1 illustrates the defense lines as they correspond to the plant states.

The first defense line (DL1) does not include plant functions. It minimizes potential for PIEs to occur in the first place and minimizes potential for failures to occur in subsequent defense lines by assuring high quality and conservatism in design, construction, and operation. The second, third, and fourth defense lines (DL2, DL3, and DL4) comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The fifth defense line (DL5) involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs.

The defense lines include measures such as engineering and operational practices, plant features, and plant functions. These measures are incorporated such that:

- The normal operation of the plant is monitored and controlled such that PIEs that lead to AOOs can be mitigated before evolving into DBAs
- The consequences are limited if a DBA does develop
- Multiple defense lines are capable of independently performing the FSFs. While this means that more than one DL is capable of independently performing the FSFs for D-in-D, DL independence from all other DLs is based on how specific DLs are credited for specific fault sequences.

Table 3.1-1 provides a high-level description of the objective, and the design means and operational means for supporting the defense lines. The following is a brief description of each of the defense lines.

Defense Line 1 (DL1)

The purpose of the first level of defence is to prevent deviations from normal operation and the failure of important SSC. This is achieved through the quality measures taken to minimize potential for failures and for initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defence. These quality measures cover the design, construction, inspections, operation, use of operational experience, periodic safety reviews, and maintenance, and testing of the plant.

DL1 measures may support the basis for assumptions made in safety analyses. For example, the use of a high-quality design process and stringent equipment qualification for the most important components support the assumption that only a single failure is considered in the Conservative Deterministic Safety Analysis discussed in Chapter 15, Subsection 15.2.1.

Examples of DL1 measures include:

- The clear definition of normal and abnormal operating conditions
- Maintenance and implementation of a quality assurance program consistent with nuclear regulations and industry standards
- Application of appropriate industry standards to the design of SSC
- Adequate design margins
- Robust design processes including design verifications
- Comprehensive testing programs

- 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 DECs.

For the BWRX-300, DL4 is comprised of two subsets of functions that are designated as DL4a and DL4b functions. DL4a functions mitigate DECs that occur without core damage. DECs 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 DECs 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.

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 DECs, the design includes independent and diverse system functions to cope with the effects of common cause failure (e.g., DL4a).
3. Diverse accident monitoring instrumentation for severe accident management (e.g., DL4b) is provided.

The defence-in-depth approach is designed to include analyses of a reasonable set of CCF scenarios to provide assurance that the plant is protected against CCF phenomena. This approach is implemented using a set of CCF application guidelines to define the CCF modes that are included, how the failure modes are applied, and which assumptions can be made regarding equipment operability.

3.1.7.9 Other Approaches for Ensuring Safety

In addition to the design principles discussed above, the BWRX-300 design incorporates the following approaches to ensure safety.

3.1.7.9.1 Simplicity in Design

An implicit approach to reliability is to deploy the design with minimal complexity, with the knowledge that complexity may be required to enhance reliability or reduce the potential for human error. Where complexity is required (e.g., self-diagnostics, redundancy within the equipment in a single division), the complexity is documented and justified as necessary and appropriate for enhancing reliability, surveillance, calibration, and other required system or equipment attributes. There are tradeoffs in complexity, such as increasing the complexity by designing the system to reduce the human actions necessary for surveillance which also decreases the potential for human error, which enhances system reliability.

The BWRX-300 is specifically designed to enhance safety through simplification and reducing its dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems (these are passive features as discussed below). These safety enhancements, in combination with its reduction in scale and complexity including a reduction in total number of active SSC, simplifies operations and maintenance. Some of the simplified design features are described in Chapter 1.

3.1.7.9.2 Passive Safety Features

The design of the BWRX-300 uses passive functions that do not require external sources of power or operator actions. DL3 functions are passive to the extent that is practicable and, therefore, have significantly less reliance on supporting systems or operator actions.

Examples of the BWRX-300 passive design features include:

1. Safety Class 1 batteries are capable of powering loads for a minimum of 72 hours. The design ensures that plant safety is maintained even after battery depletion.
2. BWRX-300 utilizes natural circulation and passive natural circulation for fuel cooling and passive containment heat removal. The plant is designed with the capability to cope with decay heat for seven days using only installed systems with no reliance on significant operator actions or external resources.

The mitigation of loss-of-coolant accidents is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, reduced number, and size of RPV nozzles as compared to predecessor designs, and elimination of fluid system nozzles located below a level well above the top of active fuel to conserve inventory. The relatively large reactor pressure volume of the relatively tall chimney region provides a substantial reservoir of water above the core. This ensures the core remains covered following fault sequences involving feedwater flow interruptions or loss-of-coolant accidents without the need for active components (such as pumps). Additionally, the RPV is equipped with isolation valves attached directly to the reactor vessel for large bore piping systems to preserve reactor coolant inventory ensuring that adequate core cooling is maintained.

The application of these design concepts is described in each system design description.

3.1.7.10 Technical Acceptance Criteria

To meet the radiological acceptance criteria, derived accepted criteria are defined for the fuel pellet, fuel cladding, RCPB and containment. Deterministic safety analyses are performed to demonstrate that these criteria have been met. A description of acceptance criteria is provided in Chapter 15, Section 15.3. Details of the deterministic safety analysis are presented in Chapter 15 Section 15.3. Table 15.3-1 for AOOs and 15.3-2 for DBAs.

3.1.8 Practical Elimination

Consistent with CNSC REGDOC-2.5.2 Section 7.3.4 (Reference 3.1-1) and IAEA SSR-2/1(Reference 3.1-5), the BWRX-300 design is such that fault sequences that could lead to an early or large radioactive release are practically eliminated.

The definition of early and large radioactive release (from IAEA SSR-2/1) (Reference 3.1-5) in this context are:

1. An early radioactive release is a release of radioactive material for which off-site protective actions would be necessary but would be unlikely to be fully effective in due time.

2. A large radioactive release is a release of radioactive material for which off-site protective actions that are limited in terms of lengths of time and areas of application would be insufficient for the protection of people and of the environment.

Fault sequences with early or large releases could be considered to have been practically eliminated if either of the following apply:

- It is physically impossible for the accident sequence to occur.
- The fault sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

Practical elimination is considered to refer only to those fault sequences leading to or involving core damage (e.g., a severe accident) for which the confinement of radioactive materials cannot be reasonably achieved.

The aim of the practical elimination concept is to reinforce D-in-D by focused analysis of those conditions having the potential for early radioactive release or a large radioactive release.

The justification of practical elimination preferably relies on a demonstration of physical impossibility for the accident sequence to occur. If this is not achievable, a demonstration of an extremely low likelihood of occurrence with a high level of confidence is provided. Sufficiently robust arguments and evidence are used to demonstrate the reliability of the lines of defence. If additional features are identified that prevent accidents or mitigation accident consequences, these features are considered for implementation as far as practicable.

The set of individual fault sequences that might lead to an early radioactive release or a large radioactive release are grouped to form a limited number of bounding cases or type of accident conditions.

Severe accident phenomena based on operating experience with predecessor advanced light water reactors serve as a starting point for consideration for practical elimination. Analyses demonstrating practical elimination are described in Chapter 15, Appendix 15A.

3.1.9 Safety Margins and Avoidance of Cliff-Edge Effects

A cliff-edge effect is described as a small change of conditions that may lead to a significant increase in the severity of consequences per CNSC REGDOC-3.6 (Reference 3.1-7).

In the BWRX-300 Safety Strategy, the principle of multiple physical barriers to the release of radioactive material and protection of those barriers is incorporated in the design as a DL1 measure. Margins are incorporated into the design of the physical barriers to demonstrate their capability in postulated scenarios that are more severe (by a small amount) than those in the design basis without incurring cliff-edge effects.

Conservative safety margins and sensitivity analyses are applied in safety analyses to account for assumptions and uncertainties. Additional details on the application of safety margins in Deterministic Safety Analysis are described in Chapter 15, Subsection 15.5.1.1. As part of the PSA, sensitivity and uncertainty analysis is conducted to demonstrate consideration of potential cliff-edge effects. (See Chapter 15, Subsection 15.6.1).

3.1.10 Design Approaches for the Reactor Core and for Fuel Storage

3.1.10.1 Design Approach for Reactor Core

The reactor core is designed to maintain the integrity of the fuel and the fuel cladding. The fundamental safety functions of control of reactivity, removal of heat from the reactor and fuel, and confinement of radioactive materials are inherent design features for the reactor core.

The reactor core, the fuel, and fuel assemblies, including fuel channels and control blades, are designed such that the reactor can be shut down, cooled, and held subcritical with adequate margin in operational states, DBAs, and DECs. Reactivity control ensures shutdown margin for shutdown states and any credible changes in core configuration. The design ensures that the fission chain reaction is controlled during operational states. The design limits positive reactivity through inherent neutronic and thermal-hydraulic characteristics, means of shutdown, and control to protect the reactor pressure boundary and prevent fuel damage.

The reactor core (including associated structures and cooling systems) is designed to withstand static and dynamic loading and vibration, to be compatible with expected chemicals, and to meet thermal material and radiation damage limits.

The reactor core design also provides for certain operator actions in accident scenarios to maintain the reactor in a shutdown condition, such as actions that might be addressed in emergency operating procedures or severe accident management guidelines.

3.1.10.2 Design Approach for Fuel Handling and Storage

The design of fuel handling and storage systems is consistent with the D-in-D approach applied to the reactor core with slightly different fundamental safety functions.

The design approach is to identify fuel handling and storage SSC that are necessary to fulfill the following fundamental safety functions for all plant states:

- Maintaining subcriticality of the fuel
- Removal of the decay heat from irradiated fuel
- Confinement of radioactive material, shielding against radiation as well as limitation of accidental radioactive releases

The Safety Strategy principle for fuel handling and storage is to leverage design and safety features in relation to fuel handling and storage that have been proven either in predecessor BWR applications or are based on operating experience.

Subcriticality is maintained by preventing criticality through use of geometrically safe configurations. The design of fuel storage systems considers the use of physical means or physical processes to increase subcriticality margins in normal operation to avoid reaching criticality during PIEs, including those PIEs arising from the effects of internal hazards and external hazards.

Fuel handling and storage systems are designed to maintain adequate fuel cooling capabilities for irradiated fuel ensuring that the fuel cladding temperature limits and/or the coolant temperature limits, as defined for operational states and accident conditions, are not exceeded.

The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed:

- With a capability to permit appropriate periodic inspection and testing of components safety features,
- With suitable shielding for radiation protection,
- With appropriate containment, confinement, and filtering systems,

- With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and
- To prevent significant reduction in fuel storage coolant inventory under accident conditions.

Appropriate systems are provided in fuel storage and radioactive waste systems and associated handling areas:

- To detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and
- To initiate appropriate safety actions

Refer to Chapter 9A, Section 9A.1 for a detailed description of the Fuel Handling and Storage Systems.

3.1.11 Considerations of Interactions Between Multiple Units

Operating experience has demonstrated that interactions or shared equipment between multiple units can cause problems for the plant and for personnel. Lessons learned include:

- Significant interactions between multiple co-located radiological sources (e.g., reactor units, spent fuel pools, or dry fuel storage facilities) could result due to concurrent or consequential initiators.
- The timing of concurrent accident sequences involving multiple radiological sources on a site can challenge shared SSC, as well as resources available for severe accident management and emergency response to the event.

Site evaluations would address multiple reactors or other co-located facilities and determine if these need to be treated as external hazards (e.g., external radiation sources) in the design of the BWRX-300. See Chapter 2, Subsection 2.2.5 for more details.

Each BWRX-300 unit would have its own safety class systems and its own safety features for DECs.

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

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.”

3.1-8 CNSC Regulatory Document REGDOC-2.4.2, “Safety Analysis – Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.”

3.1-9 IEC 60880, “Nuclear power plants – Instrumentation and control systems important to safety – Software aspects for computer-based systems performing category A functions,” International Electrotechnical Commission.

Table 3.1-1: Identification of Defence Levels

Level of Defence/DL	Objective	Design Means	Operational Means
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

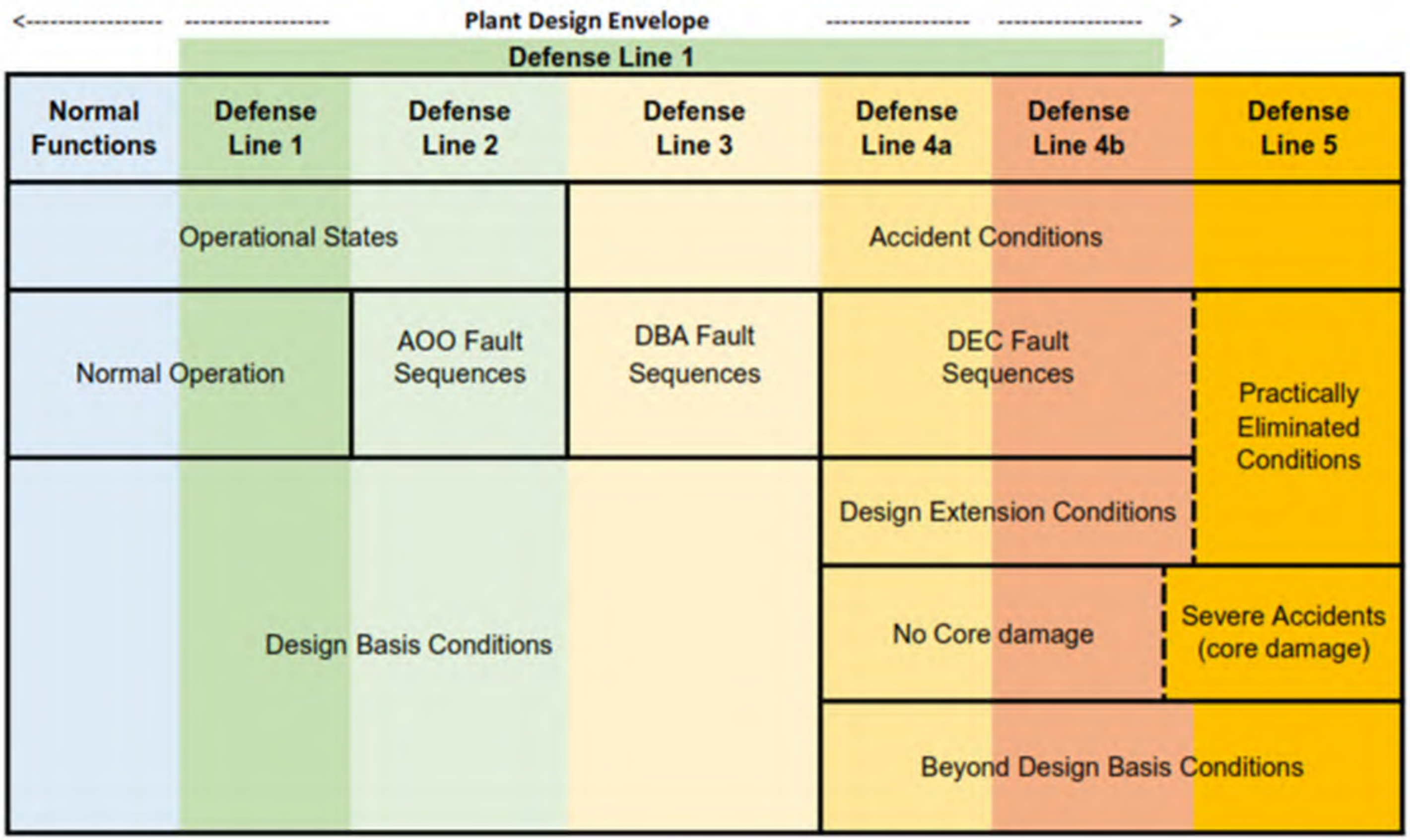

Figure 3.1-1: Defence-in-Depth – Plant States and Defense lines

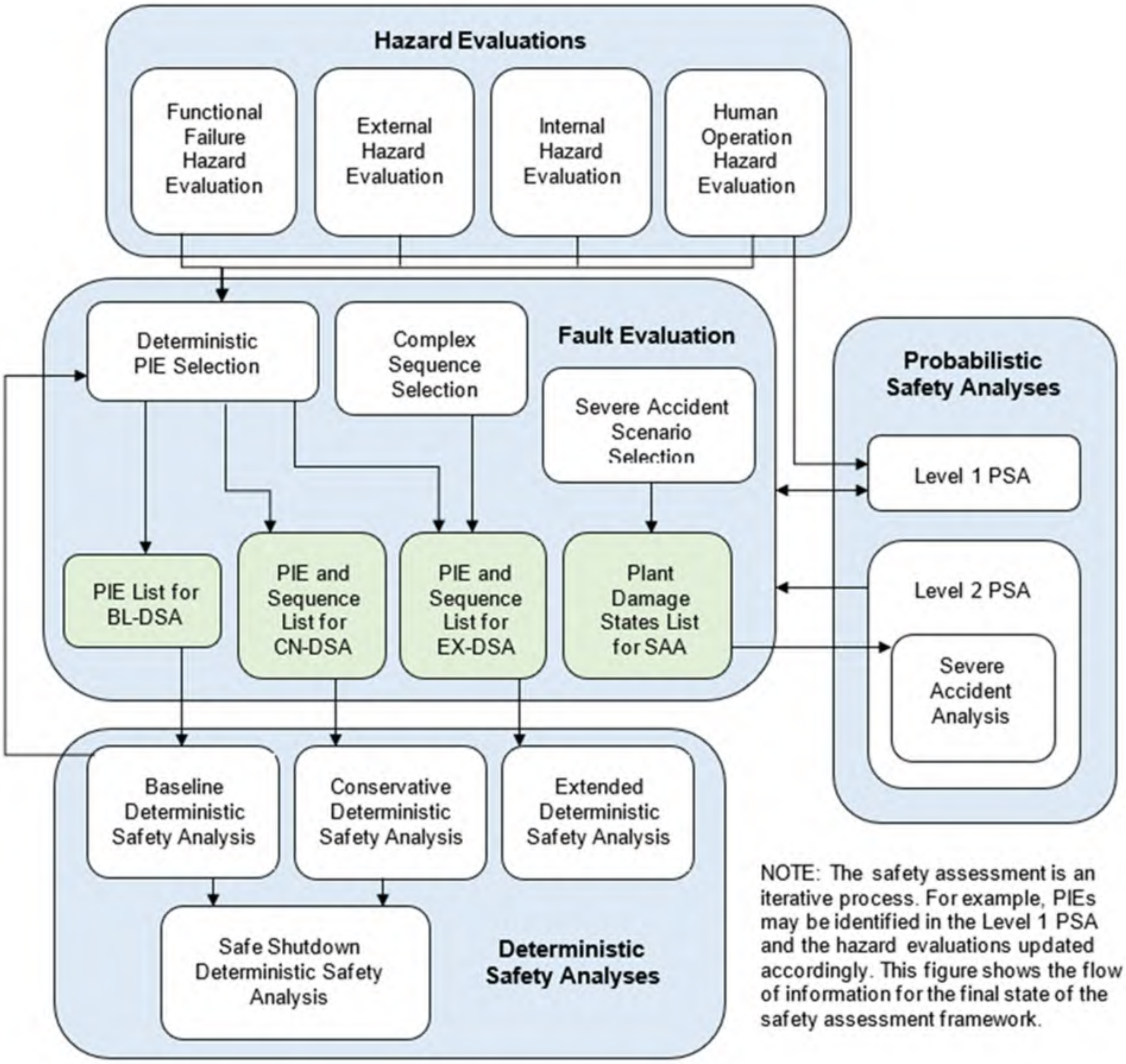

Figure 3.1-2: BWRX-300 Safety Strategy Implementation Process

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 SCC with respect to safety.

This section described how BWRX-300 SSC are classified by:

- Safety Class (SC)
- Seismic Category
- Quality Group

Classification of SSC provides a means for applying appropriate design requirements and establishes a graded approach in the selection of materials, and application of codes and standards used in design, manufacturing, construction, testing and inspection of individual SSC. Sections 3.5 through 3.8 describe the codes and standards applicable to civil, mechanical, I&C, and electrical SSC based on classification.

The classification of SSC also determines the degree of redundancy, diversity, separation, and reliability/availability required as described in Subsection 3.1.7. The requirement for environmental qualification is based on the classification of SSC as described in Section 3.9. In addition, SSC classification informs procurement and quality assurance requirements as discussed in Chapter 17.

3.2.1 Safety Classification Background

The BWRX-300 approach to classifying SSC by safety class is based primarily on deterministic methods and is directly traceable to the safety functions performed by the SSC. This approach aligns with CNSC REGDOC-2.5.2, Section 7.1, as it reflects:

- Consequences of the SSC failure to perform its safety functions
- Expected frequency of the SSC being called upon to perform its safety functions
- Time following a PIE at which, or the period for which, the SSC may be called upon to perform a safety function

A fundamental element of the BWRX-300 SSC classification approach is the direct correlation between the Defense Line in which an SSC performs a function, and the relative safety importance of that function. Functions are categorized into three safety categories, Safety Category 1, Safety Category 2, and Safety Category 3, with Safety Category 1 being the most important.

3.2.1.1 Primary Function Categorization

Primary functions are those that directly perform the FSFs in support of DL2, DL3, DL4a or DL4b. Safety Categories are applied to the primary functions as follows:

1. Safety Category 1 is assigned to DL3 primary functions. DL3 functions assure the integrity of the barriers to release, place and maintain the plant in a safe state, and provide independence and diversity for all DL2 and DL4a functions caused by a single failure (and many CCFs). Accordingly, DL3 primary functions are the most important from a safety standpoint.
2. Safety Category 2 is assigned to DL4a primary functions. Both DL2 and DL4a provide a redundant means to address PIEs (generally independent of DL3 functions) and are therefore important from a safety standpoint, although less important than DL3 functions.

DL4a functions are a backup to DL3 functions, in the unlikely event a DL3 functions fails, and therefore have a higher consequence of failure than DL2 functions and are more important from a safety standpoint than DL2 functions.

3. Safety Category 3 is assigned to DL2 and DL4b primary functions as they are relatively the least important. DL4b functions address severe accidents, which are extremely unlikely because failure of both DL3 and DL2 or DL4a functions would have to occur. Accordingly, DL4b functions are considered relatively the least important defense line functions, despite the high consequence of failure.
4. Non-Safety Category is assigned to all other functions.

The assignment of DL4a functions to Safety Category 2, to address the low probability but high consequences of failure, and the assignment of DL4b functions to Safety Category 3, based on the extremely low probability of being called upon, is consistent with CNSC REGDOC-2.5.2, Section 7.1 (Reference 3.2-3), which provides guidance on the treatment of complementary design features called upon to mitigate DECs.

In addition to categorizing primary functions by the defense line they support, function that provide a supporting role and functions that are not immediately required following a PIE are assigned to a Safety Category as described below and summarized in Table 3.2-1.

3.2.1.2 Integral Support Functions

Integral support functions are functions that support the primary function and are required to be performed concurrently with the primary function (e.g., an HVAC system maintaining the temperature of a space or area within an acceptable range during performance of the primary function (i.e., following the initiating event) to maintain equipment in an acceptable condition).

Integral support functions are considered part of the defense line function (and therefore subject to defense line function “rules,” such as independence and diversity) and are assigned the same safety category as the primary function they support.

3.2.1.3 Make-Ready Support Functions

Make-ready support functions are continuously available online functions that maintain the primary function, or a component required to perform the primary function, in a state of readiness but are not required to be performed at the time the primary function is performed. Make-ready functions must have monitoring, such that plant operators would be alerted if the make-ready support function were lost, or the readiness of the primary function or component were compromised. For example, maintaining the temperature of a pool of cooling water within acceptable limits, with monitoring by pool temperature indication is an example of a make-ready support function.

Make-ready functions are not required at the time the primary function is performed and are not considered part of the defense line function (and therefore not subject to defense line function “rules,” such as independence and diversity). The primary function would eventually be considered unavailable if the make-ready function were compromised to the extent that the primary function might be compromised. Accordingly, make-ready functions are not required to be assigned the same safety category as the primary function. However, make-ready functions are important and are therefore assigned to safety categories as follows:

- Make-ready functions that support DL3 or DL4a functions are assigned to Safety Category 3
- All other make-ready functions can be assigned to Safety Category N.

3.2.1.4 Delayed Functions

Delayed functions are primary or support functions that are not required to be performed until sometime after the initiating event. Because there would be ample time during the event to ensure these functions are available, delayed functions are not required to be assigned the same safety category as functions required immediately after the initiating event. If the function is not needed until after 72 hours into the event (but before seven days), it can be classified as Safety Category 2 (instead of Safety Category 1), and if the SSC is not needed until after seven days into the event, it can be classified as Safety Category 3 (instead of Safety Category 1 or Safety Category 2). Delayed functions are not subject to defense line function “rules,” such as independence and diversity.

3.2.1.5 Normal Functions

Normal functions that perform an FSF during normal plant operation or that maintain key reactor parameters (e.g., reactor pressure and temperature) within normal ranges, and their integral support functions, are assigned to Safety Category 3. Make-ready functions for normal functions can be assigned to Safety Category N. If failure of a normal function would likely result in an initiating event that could challenge an FSF, the function should be assigned to Safety Category 3.

3.2.1.6 Assignment of Safety Class to Components

Safety Class is assigned to components based on the safety category of the functions they perform as follows:

- Safety Class 1 (SC1) is assigned to SSC that perform a Safety Category 1 function
- Safety Class 2 (SC2) is assigned to SSC that perform a Safety Category 2 function
- Safety Class 3 (SC3) is assigned to SSC that perform a Safety Category 3 function
- Non-Safety Class (SCN) is assigned to all other SSC

Just as with functions, a time-dependency is introduced for components that perform or support DL3 and DL4a functions. Specifically, if the component is not needed until after 72 hours into the event (but before seven days), it can be classified as SC2 (instead of SC1), and if the component is not needed until after seven days into the event, it can be classified as SC3 (instead of SC1 or SC2) because there would be ample time during the event to ensure those components are available. (See Table 3.2-2)

Functions typically have a mission time, which is the time period during which the function is required to be performed. Only SSCs that are required during the mission time of the function are required to be assigned to the safety classes discussed above.

Some component classifications are made for components that perform FSFs but may not be explicitly defined as part of a defense line function. For example:

- Components that are part of design provisions that perform a FSF, whose failure is considered “practically eliminated,” are assigned to SC1. An example is the RPV.
- Components that make up the fission product barriers are assigned to SC1.
- Components that are part of the RCPB are assigned to SC1.

The safety classification of a system is the highest safety classification of any components within the system; however, the component safety classification, and not the system safety classification, defines the design rules applied to components. Assignment of safety classifications to systems is for convenience in understanding the relative importance of plant systems.

Not all components or parts of a system are necessarily assigned to the same safety class as the system itself. For example, a process system may be classified as SC 1 because one or more of its components support a DL3 function; however, the system may also contain components that support functions associated with other defense lines or components that support no defense line functions. These components are classified in accordance with the defense line functions they support.

Appendix 3A provides a list of the BWRX-300 principal components organized by system and includes their safety classification.

Structures are assigned a safety classification based on the highest safety classification of the components they house or support, excluding components whose failure, due to loss of functionality of the structure, would result in fail-safe performance of the component's safety category function(s). Design rules and performance requirements for structures are derived from their seismic category. Seismic categorization methodology is described in Subsection 3.2.3. The seismic category assigned to a structure is commensurate with its safety classification as listed in Section 3.3, Table 3.3-1.

3.2.2 Safety Classification Process

In alignment with both IAEA and CNSC guidance, this method of classifying the safety significance of SSC is based primarily on deterministic methods because the DL functions are identified using deterministic safety analyses. The deterministic methods are complemented (where appropriate) by probabilistic methods and engineering judgment.

Design rules are then applied to SSC based on their safety classification and the DL functions they support. The safety classification process is iterative with the deterministic and probabilistic safety assessment and is maintained and updated throughout the design phase.

The following outlines the BWRX-300 classification process.

Review and Definition of PIEs – Hazard evaluations are performed (as introduced in Section 3.1.6.4.1) to identify hazards with potential to challenge an FSF. The output of these hazard evaluations are potential PIEs.

Grouping and Identification of Bounding PIEs – Potential PIEs are grouped by plant effect and occurrence frequency. Bounding or representative PIEs and fault sequences are selected for deterministic safety analyses as described in Chapter 15, Section 15.2.

Identification of Plant-Specific Safety Functions to Prevent or Mitigate the PIEs – The deterministic safety analyses are performed and updated iteratively with design activities to establish the plant-specific functions responsible for maintaining the FSFs during PIEs and fault sequences. The identification of plant-specific functions and their assignment to a Defense Line is carried out in the Fault Evaluation described in Chapter 15, Section 15.2 with traceability of each function to each PIE and PIE sequence in which it is credited.

Safety Categorization of the Safety Functions – Functions are categorized in accordance with their safety significance and role in performing FSFs. As such, each function receives a safety categorization directly based on its assignment to a DL (as described in Subsection 3.2.1 above).

Identification of SSC that Provide the Safety Functions

Plant-level requirements are created for each DL function and decomposed into system-specific functional requirements to implement the credited DL functions, consistent with the plant performance modeled in the safety analyses. These requirements are then allocated to the applicable system design description which identifies the components that support the system DL functions.

Assignment of SSC to a Safety Class Corresponding to the Safety Category

Safety Class is assigned to SSC based on the SSC's role in ensuring plant safety, and the defense line and FSF supported as described in Subsection 3.2.1.6 above.

Verification of SSC Classification

The deterministic safety analyses are maintained and updated as the plant design matures. Confirmation of SSC classification is achieved when the deterministic safety analyses models reflect the final plant design and demonstrate compliance to the analysis acceptance criteria (which include rules governing how classified equipment can be credited in each analysis case). This verification is complemented, as appropriate, by insights from the PSA.

Identification of Engineering Design Rules for Classified SSC

Engineering design rules are applied to SSC based on several factors including their SC, their DL role, their status as a pressure boundary component, their role during and following earthquakes, and their operational environment. The design rules establish the scope of codes and standards applied to an SSC, as well as requirements for reliability, diversity, redundancy, and independence applicable to an SSC. These design rules are discussed in Subsection 3.1.7.

3.2.3 Seismic Categories

Seismic Category reflects SSC requirements during and after a seismic event and governs how the SSC is seismically designed and qualified. Seismic Category is assigned based on the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13 (Reference 3.2-3), and CSA N289.1, Clause 5.2.5.2 (Reference 3.2-4) as follows:

1. **Seismic Category A/B -** DL3 functions are credited with remaining operable during and after a seismic event associated with a Design Basis Earthquake (DBE) as defined in Section 3.3.1. Accordingly, SSC that perform or support DL3 functions are categorized as Seismic Category A for passive SSC or Seismic Category B for active SSC. Other SSC that are classified as SC1 per Subsection 3.2.1.6, are categorized as Seismic Category A or B. Any other SSC that are a significant contributor to PSA risk for seismic events are categorized Seismic Category A or B.
2. **Seismic Category RW-IIa** - SSC for management and storage of radiological material that, if released would exceed the dose limits defined in CNSC REGDOC-2.5.2, Section 4.2.1, are categorized as Seismic Category RW-IIa per guidance in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.143, (Reference 3.2-7). These RW-IIa SSC are seismically qualified for one-half of the site-specific DBE This approach is in accordance with CNSC REGDOC-2.5.2, Section 7.13.1, which permits the use of ASCE/SEI 43 (Reference 3.2-8) graded approach for the seismic classification of SSCs with justification. Based upon the consequences of failure, one-half of the site-specific DBE is justified as it would bound the ground motion spectra for seismic categories identified in ASCE/SEI 43 (Reference 3.2-8) for SSCs used for handling and storage of highly radioactive materials. This justification is described in NEDC-33974P (Reference 3.2-18).
3. **Non-Seismic** - All other SSC are categorized as Non-Seismic and are designed based on applicable non-nuclear requirements, such as those stipulated in the National Building Code of Canada (Reference 3.2-19).

The BWRX-300 Containment and the Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. These two structures are therefore categorized as Seismic Category A structures in the BWRX-300 design per Clause 5.2.5.2 of CSA N289.1 (Reference 3.2-4).

3.2.3.1 Seismic Interaction

SSC that are not Seismic Category A or B but whose failure during a seismic event could adversely affect the ability of any Seismic Category A or B SSC to accomplish its safety function are evaluated for seismic interaction to demonstrate that these SSC:

- Will not collapse or collide with the Seismic Category A and Seismic Category B SSC and will maintain their stability during a DBE or other relevant extreme external hazard event; or
- Impact loads that result from collapse or collision on the Seismic Category A and Seismic Category B SSC are either negligible or smaller than those considered in the design.

In accordance with requirements of Clause 7.2.1.2 of CSA N289.3 (Reference 3.2-6) and Section 6.0 of NEDO-33914 (Reference 3.2-9), interaction evaluations are performed of the Power Block structures and foundations adjacent to the Seismic Category A RB, as described in Subsection 3.3.1.2.8, to ensure:

- These structures and foundations do not collapse to compromise the safety functions of those SSC that are required to remain functional following a DBE or design tornado level event for the first 72 hours.
- The CB structure, which includes the Main Control Room (MCR) does not collapse and result in incapacitating injury to the main control room occupants or prevent their egress to the RB.

Table 3.3-1 in Section 3.3 lists the seismic category and seismic interaction evaluation requirements for structures..

Evaluations for seismic interaction of systems and components is conducted as the design advances and details supporting these evaluations are available.

3.2.4 Quality Group

In alignment with CNSC REGDOC 2.5.2, Section 7.7 (Reference 3.2-3), BWRX-300 pressure-retaining components are designed to ensure they are protected against overpressure conditions, and are classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. The selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. CNSC REGDOC-2.5.2 points to ASME Boiler and Pressure Vessel Code (BPVC) (Reference 3.2-11) to meet the requirements of different classes of pressure-retaining systems, components, piping and their supports.

BWRX-300 design utilizes a Quality Group designation per the guidance in USNRC RG-1.26 (Reference 3.2-10) as a method for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of the component. Items are classified as Quality Group A, B, C or D. The guidance and classification method are used with some clarification based on the unique design of the BWRX-300.

Table 3.2-3 tabulates the design and fabrication requirements for each Quality Group. For mechanical equipment that does not fall within the scope of USNRC RG 1.26 (Reference 3.2-10), 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.

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	• Primary and Integral support functions required within the first 72 hours of an event			
2	• Primary and integral support functions required after 72 hours but before 7 days after an event	• Primary and integral support functions required within the first 7 days of an event		
3	• Primary and integral support functions required after 7 days after an event • Make-ready support functions	• Primary and integral support functions required after 7 days • Make-ready support functions	• All primary and integral support functions	• Normal functions that perform a fundamental safety function • Normal functions that maintain key reactor parameters (e.g., pressure and temperature) within normal ranges • Integral support functions
N			• Make-ready support functions	• Make-ready support functions

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	• SSCs required within first 72 hours of event				• Components that are part of design provisions that perform a FSF, whose failure is considered “practically eliminated” • Components that make up the fission product barriers • Components that are part of the reactor coolant pressure boundary
2	• SSCs required after 72 hours but before 7 days	• SSCs required within first 7 days of event			
3	• SSCs required after 7 days	• SSCs required after 7 days	• All SSCs		
N				• All SSCs	

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(4)	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 (1)
	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(2)	API 620 or equivalent3	API 650 AWWA D100-11 ASME B96.1 or equivalent(3)	—	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

- 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 seismic categorization reflects SSC's functional and performance requirements during or after a seismic event and impacts their design.

Following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, CSA N289.1 (Reference 3.3-3) and U.S. NRC RG 1.208 (Reference 3.3-4), Seismic Category A and Seismic Category B SSC are seismically qualified to withstand the effects of a Design Basis Earthquake (DBE) that is developed:

1. Based on the geological, seismological, and geotechnical conditions at the site described in Chapter 2, Section 2.7
2. Following the performance-based approach of ASCE/SEI 43 (Reference 3.3-5) Section 2 for development of DBE for seismic design of structures achieving a target performance goal of 1E-5 per year
3. Meets the minimum earthquake requirements of CSA N289.3 (Reference 3.3-6), Clause 4.2

The development of the 5% damped Acceleration Response Spectra (ARS) defining the amplitude and frequency content of the bounding site-specific DBE input ground motion used for the seismic qualification of Seismic Category A and B SSC is discussed in Subsection 3.3.1.1.

Table 3.3-1 provides the seismic categorization of BWRX-300 structures. Per Subsection 3.2.3, the containment, and Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. As a result, the integrated RB structure, which consists of the RB, containment and containment internal structures is the only structure categorized as Seismic Category A in the BWRX-300 design. As shown in Table 3.3-1, the seismic design of the Seismic Category A structures considers Limit State LS-D response defined in Table 1-2 of ASCE/SEI 43 as essentially elastic response without any significant permanent deformation. According to U.S. NRC RG 1.208, this ensures a consistent level of safety from earthquake-caused failures defined by level of response resulting in an onset of significant inelastic deformations with a probability of unacceptable performance:

- Less than 1% for a DBE ground motion level
- Less than 10% for ground motion with 1.5 times the DBE intensity

The Radwaste Building (RWB) which processes and houses liquid, solid and gaseous radwaste is categorized as Seismic Category RW-IIa as shown in Table 3.3-1. The remaining BWRX-300 Power Block structures, which consist of the Control Building (CB), Turbine Building (TB) and Reactor Auxiliary Bay (See Chapter 1, Appendix A, Figure A1.4-1) are categorized as Non-Seismic.

Due to their proximity to the Seismic Category A RB, the RWB, CB, TB and Reactor Auxiliary Bay are evaluated for interaction with the integrated RB structure per the requirements in SSR-2/1 (Reference 3.3-7), Section 5.19, as discussed in Subsection 3.2.3.1. The interaction evaluation methodology is presented in Subsection 3.3.1.2.8. Table 3.3-1 summarizes the seismic design basis for the BWRX-300 structures based on their seismic categories. Per Table 3.3-1, the RW-IIa structures are designed per CSA N291 and U.S. NRC RG 1.143 (Reference 3.3-8), while Non-Seismic Category structures are designed in accordance with the National Building Code of Canada (NBC) (Reference 3.3-9). The primary focus of this section is on the seismic qualification of Seismic Category A and Seismic Category B SSC. The seismic design of the RW-IIa and Non-Seismic Category structures is further discussed in Chapter 9B, Section 9B.3.

Seismic robustness of Seismic Category A structures is evaluated for a Design Extension Condition (DEC) Checking Level Earthquake (CLE) as described in Subsection 3.5.6.1.

The BWRX-300 design considers Operating Basis Earthquake (OBE) and Site Operating Earthquake loads as 1/3 of the DNNP site-specific DBE. Per Appendix S to 10 CFR 50 (Reference 3.3-10), design load combinations that consider OBE and Site Operating Earthquake loads are not required, except for the design of metal containment components where the OBE loads are considered for post-flooding condition and cyclic loading considerations, as noted in Table 9B-1 in Chapter 9B. OBE is not used as reference earthquake for the BWRX-300 DNNP plant shutdown.

The DNNP BWRX-300 seismic instrumentation is discussed in Subsection 3.3.1.5. As described in Subsection 3.3.1.5, the criteria for seismic instrumentation, plant shutdown, evaluation and inspection are in accordance with the guidelines of CSA N289.5 (Reference 3.3-11) and Clause 6.5 of CSA N289.1.

3.3.1.1 Bounding Seismic Design Parameters

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the design of the BWRX-300 is based on DNNP site-specific geotechnical and seismic inputs. Bounding seismic design parameters are developed based on the data that was available prior to the completion of the characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative site-specific seismic inputs adequately address uncertainties related to the use of incomplete (preliminary) characterizations of the DNNP geotechnical and seismic conditions.

The 5% damped spectra defining the magnitude and frequency content of the DNNP bounding site-specific design ground motion are developed based on the results of probabilistic Site Response Analysis (SRA) presented in Subsection 3.3.1.1.2 using as input the dynamic subgrade properties dynamic subgrade properties described in Subsection 3.3.1.1.1.

The results of the probabilistic SRA are also used for the development of bounding stiffness and damping properties of subgrade materials that are compatible with the free-field strains generated by a typical design level earthquake event.

The bounding DBE ground motion response spectra in Subsection 3.3.1.1.3 and the bounding strain-compatible dynamic subgrade profiles discussed in Subsection 3.3.1.1.6 provide a conservative seismic design that adequately address the aleatory variabilities and epistemic uncertainties in the geotechnical properties of the DNNP site.

Five sets of ground motion time histories compatible to the bounding DBE ground motion response spectra are developed, as described in Subsection 3.3.1.1.4, for use as input for the linear seismic Soil-Structure Interaction (SSI) analysis.

3.3.1.1.1 Bounding Dynamic Subgrade Properties

The bounding seismic design parameters are developed using dynamic properties for the subgrade rock, in-situ soil, and engineered fill that are determined based on the data obtained from multiple geotechnical investigations that were completed at the vicinity of the DNNP site prior to the geotechnical site investigations and laboratory tests described in Chapter 2 Section 2.7.3.

For use as input for the probabilistic SRA described in Subsection 3.3.1.1.2, bounding subgrade dynamic profiles are developed reflecting anticipated as-built conditions at the site after construction of the BWRX-300 SMR that include compacted fill from about elevation 80 to 82 m Canadian Geodetic Datum (CGD) to the final grade at elevation 88 m CGD. The layering of the in-situ soil materials is determined based on the stratigraphy obtained from the studies presented in:

- NK054-REP-01210-00098 (Reference 3.3-12) providing data from multiple borings near the proposed BWRX-300 SMR site (B-104, B-113, B-116, and B-118), and

- 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 σTOR of ±1 m
2. The variation in the rock layers assumes ±2 m
3. The σμ ln represents the epistemic uncertainty for estimating LR (10th percentile) and UR (90th percentile) profiles
4. The σlnVs represents the aleatory uncertainty for randomization of the shear wave velocities.

Epistemic uncertainty in the distribution of the shear wave velocity profiles (σμ ln) was estimated based on the range of Vs 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 σμ ln values, a σμ 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 σμ ln value was not justified by the site data. Aleatory uncertainty considers a standard deviation for the natural log of the shear wave velocity (σlnVs) 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).

The uncertainty in the estimates of soil *Vs* is considered using a σµ 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 σµ ln values generally cover the range of estimated *Vs* 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 *Vs* for the N60 values, per the 2012 PEER Report 2012/08 (Reference 3.3-17). The selected *Vs* 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 σµ ln of 0.40 was selected. The selected σµ ln value is considered reasonable due to the limited information on the fill materials. A σlnVs value of 0.25 is used for the fill and upper till and a value of 0.15 is used for the deeper in-situ soil layers.

The BE, LR, and UR variations of the kappa parameter, used to establish consistent damping ratios for the rock layers at the site are presented in Table 3.3-4. The kappa value was estimated following the guidance of the 2013 EPRI TR-1025287 (Reference 3.3-16) for CEUS firm rock profiles with a thickness of less than 1000 m and a total standard deviation of 0.47 for kappa based on the 2014 PEER Report No. 2014/12 (Reference 3.3-18).

The BE, LR and UR of the shear wave velocity profile representing the assumed as-built conditions are presented in Figure 3.3-1.

The dynamic subgrade stiffness properties of in-situ soil and engineered fill materials in Table 3.3-3 correspond to small-strain levels. To account for the nonlinearity of the engineered fill and in-situ soil materials. The following two sets of strain-dependent property curves are recommended in EPRI TR-1025287 (Reference 3.3-16, Section B-3.3):

- EPRI curves from the 1993 EPRI TR-102293, "Guidelines for determining design basis ground motions (Reference 3.3-19)
- Peninsular Range curves, Silva, W.J., N. Abrahamson, G. Toro and C. Costantino. (1996). Description and validation of the stochastic ground motion model (Reference 3.3-20)

The Peninsular Range curves are used for the development of bounding seismic design parameters to account for the strain-dependance of the soil and engineered fill dynamic stiffness and damping properties. The EPRI curves are not considered because the results of SRA indicated excessive softening of the soil and fill layers which can result in unconservative estimates of the seismic response at the ground surface, per the 2013 EPRI TR-1025287 (Reference 3.3-16, Section 5.0, and Figure 5-7).

3.3.1.1.2 Site Response Analyses

Probabilistic Site Response Analyses (SRA) are performed to accommodate the effects of overlying materials on the seismic hazard considering the epistemic uncertainties and aleatory variabilities in the site parameters to preserve the desired hazard levels and performance goals per requirements of CSA N289.2 (Reference 3.3-21) and regulatory guidelines of U.S. NRC RG 1.208. These SRA consider as-built conditions at the DNNP site after the excavation, construction, and backfilling. The equivalent linear approach is used for the SRA to account for the non-linear response of the soil. Curves representing the shear modulus reduction (G/Gmax) 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 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/Gmax) and damping of the soil materials with strain are randomized into 60 realizations with correlated log-normal distribution using the Darendeli model (Reference 3.3-23). The damping of subgrade materials is limited to 15% in accordance with the regulatory guidance of ASCE/SEI 4 (Reference 3.3-24), Section C5.2 and U.S. NRC RG 1.208, Appendix E. Figure 3.3-4 shows examples of randomized modulus reduction and material damping curves. The thick black lines in these plots designates the resulting mean curves.

Approach 1, from the approaches defined in NUREG/CR-6728 (Reference 3.3-25), is implemented for the SRA, where the reference site Uniform Hazard Response Spectra (UHRS) with Mean Annual Probability of Exceedance (MAPE) of 1E-3, 1E-4 and 1E-5, are directly used as input control motions and propagated from the bedrock with reference shear wave velocity of 2,800 m/sec through the randomized subgrade profiles. This allows the 5% damped ARS results of Approach 1 SRA to be directly used for the development of the UHRS representing the seismic hazard at the horizons of interest.

Approach 1 is selected as appropriate approximation for the purposes of development of bounding seismic parameters using a preliminary site information.

The reference site UHRS at 1E-03, 1E-04, and 1E-05 MAPE levels are developed using the results of the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26). Between the different options considered in this PSHA, Option 2 for CAV filtering of magnitudes 5 and above is used as input for the Approach 1 SRA, as it provides the greater seismic hazard. Figure 3.3-5 shows the bedrock UHRS used as input for the SRA.

Using the random vibration theory, power spectral density functions for the reference site motions are calculated iteratively from the input UHRS and propagated throughout the randomized shear wave profiles to calculate power spectral density functions at the horizons of interest. 5% damped ARS at each horizon of interest are then calculated from their corresponding power spectral density functions implementing the random vibration theory approach.

For each of the 9 base cases shown in Figure 3.3-2 and MAPE considered, log-mean (μ_i) and log-Standard Deviation (σ_i) 5% damped ARS results are calculated form 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 σ_T and σ_{Ep} are calculated as follows, representing the composite (total) uncertainty and epistemic uncertainty of the calculated hazard at the horizons of interest, respectively:

$$\sigma_T = \sqrt{\sum_i w_i \left((\mu_i - \mu_T)^2 + \sigma_i^2 \right)}$$

$$\sigma_{Ep} = \sqrt{\sum_i w_i (\mu_i - \mu_T)^2}$$

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 (σ_{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).

The following horizontal and vertical spectra define the amplitude and frequency content of the DNNP site-specific DBE ground motion for the SSI analysis of the BWRX-300 deeply embedded RB structure:

1. Foundation Input Response Spectra (FIRS) defining the DBE ground motion at bottom of RB Foundation.
2. Performance Based Surface Response Spectra (PBSRS) defining the DBE ground motion at the finished plant grade elevation.
3. Performance Based Intermediate Response Spectra (PBIRS) defining the DBE ground motion at intermediate embedment depth elevation established, following the guidelines in NEDO-33914 Revision 2, Section 5.2.2 at the top of the rock elevation having a significant contrast between rock and overlaying soil shear wave velocities.

The purpose of PBIRS is to ensure the ground motions used as input for the SSI analyses of deeply embedded structures are adequate throughout the depth of the embedment.

Horizontal FIRS, PBSRS and PBIRS are developed following the performance-based approach criteria of ASCE/SEI 43, Section 2 for DBE with a target performance goal of 1E-5. Instead of using UHRS representing the mean estimate of the seismic hazard as mandated by ASCE/SEI 43, the bounding FIRS, PBSRS and PBIRS are conservatively developed using the 1E-4 and 1E-5 MAPE UHRS representing UB estimates of the seismic hazard. These UB UHRS are developed as described in Subsection 3.3.1.1.2 to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology. The resulting horizontal Ground Motion Response Spectra (GMRS) are further adjusted to meet the minimum required response spectra requirement using the generic spectrum in CSA N289.3, Clause 4.3.2 anchored at the minimum peak ground acceleration value of 0.1g.

Horizontal reference site hard rock GMRS is also developed following the ASCE/SEI 43 performance-based approach using the UHRS obtained from the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26)representing the reference site hazard with MAPE of 1E-4 and 1E-5. This reference site hard rock spectrum is used to conservatively neglect the de-amplifications of the reference hazard motion as it propagates through the rock column. A single rock design ground motion response spectrum is developed as a conservative representation of the amplitude and frequency content of the horizontal rock GMRS by enveloping, as shown in Figure 3.3-10 the three GMRS representing the seismic hazard at FIRS, PBIRS and reference site hard rock horizons.

The horizontal PBSRS representing the amplitude and frequency content of the design motion at the ground surface are increased to conservatively account for the uncertainties in the soil column properties that may result in spectral peak shifts by connecting the spectral peaks in the PBSRS at frequencies of 8.3 Hz and 20.4 Hz using linear interpolation in the logarithmic space.

Figure 3.3-11 presents the development of the enveloping 5% damped PBSRS representing the amplitude and frequency content of the horizontal design ground motion at the finished grade elevation.

Vertical rock GMRS and PBIRS are developed by applying frequency-dependent Vertical-over Horizontal (V/H) ratios to the bounding horizontal spectra, in accordance with the requirements of CSA N289.3, Clause 4.3.3.3 and U.S. NRC RG 1.208.

The rock V/H ratios that are used for calculation of vertical rock GMRS, are constructed using the CEUS hard rock V/H ratios from NUREG/CR-6728 (Reference 3.3-25). The vertical PBSRS are calculated using soil V/H that are constructed following the methodology for CEUS soil sites using the procedure outlined in Appendix J of NUREG/CR-6728 (Reference 3.3-25). The rock and soil

V/H ratios used for calculation of the bounding vertical ground motion design spectra are presented in Figure 3.3-12.

Figure 3.3-13 presents the site-specific horizontal and vertical rock Design Ground Response Spectrum (DGRS) and PBSRS defining the bounding design ground motion for the seismic analysis of the BWRX-300 Seismic Category A structures and for the seismic interaction evaluations discussed in Subsection 3.3.1.2, and compares these bounding values to 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.

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 (Δt) 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.

2. In accordance with Clause 4.4.4.5 of CSA N289.3, the power spectral density of the modified ground motion history is computed as described in ASCE/SEI 4, Section 2.6.2, and shown not to have significant gaps in energy at any frequency over this frequency range.
3. The total duration of time histories has to be no less than 15 seconds with minimum strong motion duration of 6 seconds per CSA N289.3, Clause 4.4.4.2 and long enough to provide an adequate representation of the Fourier components at low frequency.
4. Time histories used as input for the seismic response analyses have a strong motion duration, and ratios V/A and AD/V^2 (where A, V, and D, are the peak ground acceleration, velocity, and ground displacement, respectively) that are consistent with those of appropriate controlling events developed using the disaggregation data from in NK38-CORR-03611-0847339 (Reference 3.3-28)
5. The set of three modified ATHs representing the ground motion in the three orthogonal directions (two horizontal and one vertical) are statistically independent. Each pair of ground motion histories is considered statistically independent when the absolute value of their correlation coefficient does not exceed 0.16, satisfying the requirement of CSA N289.3, Clause 4.4.4.6.
6. The ATHs are baseline corrected to ensure the ground velocity converges to zero at the end of the earthquake record and maintains a zero-mean value over the time history duration.

Per recommendations of NEDO-33914 Revision 2, Section 5.2.3, the time step of the modified time histories is refined to 0.0025 seconds for the purposes of calculating high frequency in-structural responses, which exceeds the requirements of CSA N289.3, Clause 4.4.4.2.

Spectral matching of the seed time histories is completed using the time domain spectral matching procedure proposed by Lilhanand and Tseng (Reference 3.3-29) and later modified by Abrahamson (Reference 3.3-20) and Al Atik and Abrahamson (Reference 3.3-31). Figure 3.3-151, Figure 3.3-16, and Figure 3.3-17 present an example comparison of the original and spectrally matched time histories for the HWA026 records matched to the target rock design ground motion response spectrum. These plots demonstrate the non-stationary characteristics of the time histories are preserved. The most noticeable changes to the time histories are due to low frequency wavelets added at later portions of the time histories.

Response spectrum of the generated acceleration time histories are computed and compared to the appropriate target response spectra. A small scaling factor is applied to the time histories to increase the spectra and meet the design criteria. Finally, the cross-correlation coefficients, peak values, Arias Intensity, and Power Spectral Density function are computed for the spectrally matched time histories.

Figure 3.3-18 presents the normalized Arias Intensity, and the power spectral density function for the horizontal HWA026 components that are spectrally matched to the rock design ground motion response spectrum. Figure 3.3-19 presents the response spectra for spectrally matched horizontal and vertical components of record HWA026.

3.3.1.1.5 Percentage of Critical Damping

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, damping values assigned to the structures and components in the SSI analysis model are in accordance with provisions of CSA N289.3, Clause 6.6, and ASCE/SEI 43, Section 3.3.3. The damping ratio values specified in Table 4(a) of CSA N289.3, Table 3-1 of ASCE/SEI 43, and U.S. NRC RG 1.61 (Reference 3.3-32) are used to represent the dissipation of energy in different elements. Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, lower (Response Level 1) damping ratios are used for generating in-structure demands for qualification of equipment and systems. The higher (Response Level 2) damping values can be used for development of seismic demands for structural design per ASCE/SEI 43, Section 3.3.3 and U.S. NRC RG 1.61, Section C.1.2, respectively.

The damping properties assigned to soil materials in the SSI analysis model take into account the stress-strain properties corresponding to the level of seismic input per requirements of CSA N289.3, Clause 6.6.3. Stiffness and damping properties of subgrade materials compatible to the strains generated by design level earthquake event are developed based on results of Approach 1 SRA in Subsection 3.3.1.1. The strain-compatible damping of the subgrade materials is limited to 15% in accordance with the recommendations of ASCE/SEI 4, Section C5.2 and the regulatory guidance of U.S. NRC RG 1.208, Appendix E.

Table 3.3-6 lists damping values used in the seismic analysis of structures and components. These damping values are applicable to all modes of a structure or component constructed of the same material.

Damping values for subsystems including piping and equipment are obtained using the procedures described in Subsection 3.3.1.3.

3.3.1.1.6 Supporting Media for Seismic Category A Structures

Consistent with regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the input subgrade properties for the site-specific SSI analysis of the BWRX-300 integrated RB structure are based on the geological, seismological, and geotechnical investigations and take into account the random nature and inherent uncertainties of soil material properties.

In accordance with the regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the SSI analysis uses at least three sets of subgrade profiles representing BE, UB, and Lower Bound (LB) estimates of the subgrade material properties. These profiles are representative of the as-built conditions at the DNNP site. The LB and UB shear wave velocities and damping reflect a minimum coefficient of variation of each layer properties of ±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

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 ($V_{S_{BE}}$). The LB and UB shear wave velocity profiles are calculated as the 16th and 84th percentiles, respectively, using the following equations:

$$V_{S_{LB}} = \min\left\{e^{ln(V_{S_{BE}})-\sigma}, \frac{V_{S_{BE}}}{\sqrt{1.5}}\right\}$$

$$V_{S_{UB}} = \max\left\{e^{ln(V_{S_{BE}})+\sigma}, V_{S_{BE}} \times \sqrt{1.5}\right\}$$

   where σ is the logarithmic standard deviation and the terms $V_{S_{BE}} \times \sqrt{1.5}$ and $V_{S_{BE}}/\sqrt{1.5}$ 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 (ν) 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

5, following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, and U.S. NUREG-0800 (Reference 3.3-33), SRP 3.7.2.

The BWRX-300 one-step seismic SSI analysis approach provides demands for the seismic design and qualification of SSC for all frequencies of interest and adequately captures the effects of SSSI for the integrated RB with adjacent structures and foundations. The BWRX-300 seismic analysis approach follows the guidance of NEDO-33914 Revision 2, Section 5.0 to address current limitations in U.S. NUREG-0800 SRP 3.7.2 when capturing the effects of seismic interaction of the deeply embedded RB structure with adjacent structures through the subgrade, as identified in NUREG/CR-7193 (Reference 3.3-34), Section 1.5.11.

The seismic SSI analyses are performed using the sub-structuring method in CSA N289.3, Clause 5.3.5, and ASCE/SEI 4, Section 5.4 and the ACS SASSI (a system for analyses of soil-structure interaction, see Appendix 3B) computer program to calculate the seismic response of the RB SSI system. The SSI analysis model consists of the integrated RB structure, the surrounding subgrade and the excavated volume of the subgrade materials replaced by the embedded portion of the RB structure, near field backfill materials and the models representing the dynamic properties of the foundations and structures surrounding the RB.

The sub-structuring method allows the seismic response of the SSI system to be obtained by subdividing the problem into a series of simple subproblems that can be solved separately. Using the principle of superposition, the results of different sub-analyses are combined to obtain the final solution for the SSI problem. The solution for the seismic response of the BWRX-300 RB structure, is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.

The linear elastic SASSI analyses are performed on one-step structural models that accurately represent the geometry and dynamic properties of the integrated RB structure and its interaction with the subgrade. These structural models have a refined FE mesh that is identical to the mesh of the models used for the static analyses, and that can transmit the entire frequency range of interest for the seismic design of the RB SSC. These models assume isotropic elastic material properties of structural members and surrounding subgrade and neglect any non-linearity at the soil-structure contact interfaces.

The linear elastic assumption allows a set of design and sensitivity SASSI one-step approach analyses to be performed on refined RB structural models with a large number of interaction nodes. The superposition principle, which is applicable only for linear elastic analyses, allows the SASSI stress results obtained from different dynamic and static analyses to be combined with the results of static analyses in seismic design load combinations.

Far-field interaction nodes are established at the surface of each soil layer through the RB shaft embedment depth to capture the horizontal and vertical components of the far-field motion in the SSI model. The responses calculated from these far-field interaction nodes are used to monitor the propagation of the input control motion through the RB embedment depth.

To account for the non-linear response of subgrade materials, strain-compatible subgrade properties are used that are developed based on the results of equivalent linear probabilistic SRA as described in Subsection 3.3.1.1. The uncertainties related to variation of soil and rock properties are addressed in the design of RB SSC by using seismic demands calculated as an envelope of the results obtained from SSI analysis cases of BE, LB, and UB subgrade dynamic profiles.

Input ground motion ATHs are applied to the SASSI model at the RB foundation bottom elevation as vertically propagating coherent:

- Shear waves for horizontal components of the input motion
- Compression waves for the vertical component of the input motion

The horizontal control motion is applied to the SASSI model in a manner that is consistent with the 1-D wave propagation SRA approach discussed in Subsection 3.3.1.1.

As described in Subsection 3.3.1.1, five sets of three input motion ATHs are used as input for the SSI analyses to mitigate the uncertainty in the computed responses due to the phasing of the time history frequency components.

As described in Subsection 3.3.1.2.3, uncertainties related to variations of the input SSI parameters are addressed by results of sensitivity analyses following the recommendations in Section 5.3 of NEDO-33914 Revision 2.

Frequencies of Analysis

Following the guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the frequency range considered in the seismic SSI analysis is based on the frequency content of the input ground motion, the soil properties, the building dynamic properties, including properties of the subsystems, and the response parameter of interest.

The solution for the response of the SSI system is obtained at a selected set of frequency points and then interpolated for other frequency points. The analysis is performed for a cut off frequency value established based on the largest value required by the following four criteria of ASCE/SEI 4, Section 5.3.5(b):

1. Twice the highest dominant frequency of the coupled soil-structure system or
2. The highest structural frequency of interest, or
3. The frequency at which the Fourier amplitude of input motion has passed its peak value and has reached 10% of the peak value, and
4. 20 Hz.

Criteria used to determine the highest dominant frequency and lower cutoff frequency values are described in Section 5.3.2 of NEDO-33914 Revision 2.

Sensitivity SSI analyses required to determine lower cutoff frequency values are performed for the stiffest UB subgrade profile that provides bounding responses at high frequencies.

The value of cutoff frequency determined by the criteria described above is used for the analysis of the UB subgrade profile. The analyses of the softer BE and LB profiles may use lower values for the cutoff frequency. In this case, it shall be demonstrated that the analysis of the UB profile provides responses that are bounding for frequencies higher than the cutoff frequencies used for the analyses of the softer subgrade profiles by comparing transfer function and 5% damped In-Structure Response Spectra (ISRS) results for responses at key locations within the building, selected as described in Subsection 3.3.1.2.5.

The frequencies of analysis are selected at sufficiently small frequency intervals. Transfer function amplitude results for responses at the key locations, selected as described in Subsection 3.3.1.2.5, are inspected to detect any numerical anomalies in the interpolated transfer functions (e.g., sharp narrow spikes) that can potentially affect the accuracy of results. If present, the effects of these anomalies in the interpolated transfer function results are evaluated using additional frequencies of analysis to ensure the anomalies in the transfer function interpolations do not affect the accuracy of the calculated responses.

Acceleration transfer functions and 5% damped ARS are also calculated for the response of SSI model free-field interaction nodes to check the amplitude and frequency content of the in-column free-field motion throughout the RB embedment depth.

3.3.1.2.2 Procedures Used for Analytical Modeling

SSI analyses of the integrated RB structure, which is primarily constructed of Steel Bricks™ as described in Subsection 3.5.1, are performed on 3-D FE models that meet the structural modeling requirements of CSA N289.3, Clauses 5.3.2 and 6.2, and ASCE/SEI 4, Section 3.

In addition to the integrated RB structures, simplified models of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and their foundations are included in the model to capture the SSSI effects in the RB seismic design.

Dynamic Finite Element Modeling of Integrated RB Structure

In accordance with requirements of Clause 6.10.4 of CSA N291, U.S. NUREG-0800, SRP 3.7.2, Subsection III.3.D, and ASCE/SEI 4, Section 3.4.2, the integrated RB structural FE model represents all mass expected to be present at the time of the earthquake including mass due to:

- Weight of the structure
- Weight of permanent equipment
- Mass equivalent to floor load of 2.4 kPa for miscellaneous dead weights such as minor equipment, piping, and raceways
- Weight of building elements not represented in the structural model (e.g., secondary members, siding partitions)
- Expected live load, not less than 50% of the live load specified for the design
- At least 25% of the specified design snow loads

The dynamic FE model also includes the inertia associated with the hydrodynamic effects of the fluids contained in various pools inside the RB and tanks in the RWB. The hydrodynamic effects that consist of the impulsive and convective (or sloshing) components are considered in accordance with the requirements of Clause 6.9 of CSA N289.3, Section 3.6.3 of ASCE/SEI 4, and Chapter 5 of ACI 350.3 (Reference 3.3-35). The hydrodynamic mass is included in the model by.

- Distributing the horizontal impulsive fluid mass over the pool and tank walls that are perpendicular to the direction of motion in accordance with the guidelines in ACI 350.3
- Lumping the entire vertical fluid mass on the pool slab or tank bottom.

The convective (sloshing) component of the hydrodynamic mass is not explicitly included in the global analysis model since its contribution is small and is associated with very low frequencies insignificant for the overall response. To account for the sloshing hydrodynamic effects, the design considers quasi-static sloshing pressure loads applied on the pool and tank walls in accordance with Section 9.4 of ASCE/SEI 4.

Beam and shell elements are used to adequately represent the configuration of all main structural members in the integrated RB. The FE model includes gross discontinuities such as large openings and member eccentricity. Thick shell elements are used to model the Steel Bricks™ shear walls, slabs, and mat foundation. 3-D beam elements are used to model the steel columns, beams, and trusses. The shell and beam elements are established at the centreline of the wall, slab, beam, column, and truss elements. Rigid beam and shell elements are used to model 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:

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:

$$[M] \{ \ddot{u}(t) \} + [C] \{ \acute{u}(t) \} + [K] \{ u(t) \} = \{P(t)\}$$

Where:

$\{ u(t) \}$	=	time dependent displacement of nonsupport points relative to the supports.
$\{ \acute{u}(t) \}$	=	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 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 Δt 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 Δt 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 ±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

$$L \leq \frac{\lambda}{4} = \frac{\nu}{4f} = \frac{\nu T}{4}$$

where: λ and ν are the wavelength and wave velocity, respectively.

The frequency f, or period T, correspond to the cutoff frequency of Subsection 3.3.1.3.4.

Modeling of Equipment

For dynamic analysis, Seismic Category A and B equipment is represented by lumped mass systems which consist of discrete masses connected by weightless beam elements and/or by any other appropriate finite element representation. The criteria used to lump the masses are:

A. The number of modes of dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than the cutoff frequency specified in Subsection 3.3.1.3.4.

B. Mass is lumped at any point where a significant concentrated weight is located.

C. For equipment with a free-end overhang span whose flexibility is significant compared to the centre span, a mass is lumped at the overhang span.

D. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system.

Modeling of Piping Systems

Mathematical models for Category A and B piping systems are constructed to realistically reflect the dynamic characteristics of the system. The continuous system is modeled as an assemblage of pipe elements (straight sections, elbows, and bends) supported by hangers and anchors, and restrained by pipe guides, struts, and snubbers. Pipe and fluid masses are lumped at the nodes and connected by the weightless elastic beam elements which reflect the physical properties of the corresponding piping segment. The mass node points are selected to coincide with the locations of large masses, such as valves, pumps, and motors, and with locations of significant geometry change. All concentrated weights on the piping system, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.3.1.3.4. The torsional effects of valve operators and other equipment with off-set centre of gravity with respect to the piping centreline are included in the analytical model. The pipe length between mass points is no greater than the length with a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.3.1.3.4 when calculated as a simply supported beam with uniformly distributed mass.

Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are excluded from the piping model of the main line in accordance with CSA N289.3.

All pipe guides and snubbers are modeled to produce representative stiffness to reduce model uncertainties. Snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The stiffness of the supporting structures is included in the analysis unless the supporting structure is shown to be rigid.

Modeling of Reactor Pressure Vessel and Internals

Because of the significant dynamic interaction between the RB and RPV and internals, the latter are integrated into the RB model as discussed in Subsection 3.3.1.2.

The mathematical model of the RPV and internals consists of a LMS model connected by linear elastic members and 3D finite element models. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear.

To facilitate hydrodynamic mass calculations, mass points (e.g., representing the fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into two representative lengths. These lengths represent the longest and shortest housings to adequately represent the full range of frequency response of the housings. In order to reduce the complexity of the dynamic model, the light components (such as in-core guide tubes and housing, sparger, and their supply headers) are excluded from the RPV mathematical model. However, the dynamic response of selected components is determined from a subsystem analysis after the system response is found.

Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped inside the RPV vessel. Although the dynamic coupling between the vertical hydrodynamic masses is not considered, the vertical hydrodynamic masses themselves are properly accounted for. Dynamic loads due to vertical motion are added to, or subtracted from, the static weight of component, whichever is more conservative.

The shroud support plate is modeled as a rigid link in the translational direction since it is loaded in its own plane during a horizontal dynamic event. The shroud support legs, and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

Due to the small clearances in the horizontal directions, the fuel assembly is adequately modeled as a linear system for subsystem and system analysis. In the vertical direction, the fuel assembly has the potential to lift off from its seat and a non-linear representation is required if the vertical applied and reaction forces are sufficient to cause fuel lift. Furthermore, the interface between the fuel channel and lower plate tie plate is not rigid and a non-linear model to account for slippage may be appropriate.

The weight of asymmetric secondary components, such as attached equipment, is uniformly redistributed around the node point circle. Asymmetric equipment is modeled using finite element or LMS methods.

3.3.1.3.4 Basis of Selection of Frequencies

The cutoff frequency selected in the time history and response spectrum analyses ensures that all significant modes are included in the superposition. Higher modes which cumulatively contribute less than 10% of the total system response are not considered in the superposition of the individual modal values.

The cutoff frequency for seismic and other dynamic loads follows Subsection 3.3.1.2. For seismic load, it is estimated that all modes up to 100 Hz are included.

For all other dynamic analysis, it is estimated that the cutoff frequency will be 100 Hz, as long as no more than 5 percent of the total strain energy of the system remains beyond this cutoff frequency.

Where practical, to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are approximately less than half or more than twice the dominant frequencies of the support structure. Moreover, in any case, the equipment is analyzed or tested or both to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

3.3.1.3.5 Analysis Procedure for Damping

Damping of Primary Subsystems

Primary Subsystems consist of the RPV and internals.

Damping values for seismic analysis of primary subsystems using the Modal Superposition are presented in Table 3.3-7. These damping values are in accordance with ASCE/SEI 43 and CSA N289.3.

α, β –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, ±15 percent peak broadening is applied to the spectra to account for the RB support structure modeling uncertainties. In general, using the SRSS method to combine modal responses is conservative since the maximum modal responses due to each component of multi-support excitation do not occur simultaneously. For certain “closely spaced” support with highly correlated support excitations, the SRSS superposition may yield unconservative responses. In this case, the modal responses of the “closely correlated” supports are combined algebraically first. Then, correlated sums are combined with the contributions for uncorrelated supports using the SRSS method.

3.3.1.3.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.3.1.3.1 are satisfied.

3.3.1.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are considered in the modeling of subsystems as discussed in Subsection 3.3.1.3.3.

3.3.1.3.12 Effects of Differential Building Movements

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event.

Differential endpoint or restraint deflections cause forces and moments to be induced in the system. As discussed in Subsection 3.3.1.3.9, the stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

Refer to Subsection 3.3.1.2 for the methodology used to obtain differential displacements used in the evaluation of subsystems.

3.3.1.4 Seismic Analysis of Other Subsystems

Seismic demands for the evaluation of other subsystems are developed based on ISRS, ATHs and relative displacements calculated with the Response Level 1 structural damping values in accordance with CNSC REGDOC-2.5.2, Section 7.13.1. The use of models with higher (Response Level 2) damping values can be justified based on the level of stress response as applicable to these structures.

Per Clause 6.5.2 of CSA N289.3, the seismic input at support points for the dynamic analysis of decoupled subsystems are ISRS or time histories representing the in-structure design translational motion in the two horizontal and the vertical directions due to the three components of the input earthquake motion.

If the in-structure rotations are significant, rotational ISRS and ATHs are developed and used for the design of the decoupled subsystems. Relative displacements between different support points of subsystems with multiple or distributed supports are also considered in the evaluation.

3.3.1.5 Seismic Instrumentation

In accordance with the requirements in CNSC REGDOC-1.1.2 (Reference 3.3-39), Section 4.5.6, and CNSC REGDOC-2.5.2, Section 7.13.1, seismic instrumentation is used to monitor the seismic activity at the site for the lifecycle of the reactor facility, starting from commissioning, including outages, until fully decommissioned.

The design of BWRX-300 seismic instrumentation satisfies the more stringent requirements for large reactors in CSA N289.5, Clause 5 in addition to Clauses 1 to 3 and 8 to 10.

The handling of seismic instrumentation system data records is in accordance with requirements of Clause 10 of CSA N289.5. When required, the seismic instrumentation requirements of CSA N289.5 are augmented by the requirements of U.S. NRC RG 1.12 (Reference 3.3-40).

The required actions after an earthquake follow the provisions of CSA N289.1.

3.3.1.5.1 Location and Description of Instrumentation

Free-Field Instrumentation

In accordance with the requirements of Clause 5.2.2 of CSA N289.5, at least two triaxial accelerometers are installed outside of the structure-ground interaction influence of the Power Block, but as close as practicable to the reactor to monitor the free-field ground motion at the BWRX-300 site at the plant grade and close to the RB bottom elevations.

In accordance with U.S. NRC RG 1.12, Section C.1.2, because the deeply embedded RB is founded at a depth more than 12 m below finished grade elevation, installation of a second free-field downhole accelerometer is considered at the bottom of the RB foundation, below the free-field accelerometer at finished grade level.

Structure and Equipment Instrumentation

In accordance with the requirements of Clause 5.2.3.1.2 of CSA N289.5 and Section C.1.2 of U.S. NRC RG 1.12, triaxial accelerometers are installed at several locations inside the RB including:

- One at the top of the mat foundation
- One on the containment internal structure close to the reactor vessel
- One close to the top of the containment internal structure
- One close to the top of the containment structure
- One at the operating floor elevation

Also, in accordance with Clause 5.2.3.1.3 of CSA N289.5, three additional triaxial accelerometers are installed outside of the RB, either at locations of seismically qualified SSC or at other locations that are deemed important.

The specific locations for instrumentation are determined to obtain the most pertinent information consistent with the selected key locations in the RB model to enable easy comparison between the measured and calculated in-structure responses. The sensors are installed such that occupational radiation exposures associated with their location, installation, and maintenance are maintained as low as is reasonably achievable.

Structure and equipment instrumentation stations recording are configured to be accessible for maintenance during full-power operation in compliance with the guidance of U.S. NRC RG 1.12. For sensors installed in inaccessible areas, provisions for data recording and an external remote alarm indicating actuation are provided.

Recording and Playback Equipment

Recording and playback units are provided for multiple channel recording and playback of the triaxial accelerometer signals. Characteristics and installation requirements of the recording and playback equipment follow the guidelines in U.S. NRC RG 1.12.

Accelerometers can measure acceleration amplitudes of at least 2g in accordance with Clause 5.1.6.1 of CSA N289.5.

Power Sources

In accordance with Clause 5.1.7.2 of CSA N289.5, a dedicated standby power source is provided for the seismic instrumentation. This backup power source can provide a minimum of 6 hours of continuous operation of any accelerometer or a minimum of 24 hours of continuous operation of any accelerograph in the event of failure of all external power sources.

The central unit of the seismic instrumentation system incorporates a self-contained seismically qualified standby power source dedicated for providing the system a minimum of 6 hours of continuous operation in the event of failure of all external power sources.

3.3.1.5.2 Design and Installation

In accordance with the requirements of Clause 8 of CSA N289.5, all components of the seismic instrumentation system and their supports are designed and installed to maintain their structural integrity, and to remain operational during and following a DBE. Accessibility for servicing and recalibration, anchorage and protection from adverse conditions that can affect their performance are also considered in the design.

Prior to the installation, the operational reliability of the seismic monitoring instrumentation is demonstrated, in accordance with Section C.4.7 of RG 1.12, by using prototype, environmental, vibratory, or historical test results.

3.3.1.5.3 Maintenance and Testing

Maintenance and testing of seismic instrumentation are defined in accordance with the requirements in Clause 9 of CSA N289.5, documented before the first facility startup, and updated as necessary following any modification to the system. All components of the seismic instrumentation system are maintained and tested to ensure that a maximum number of instruments are kept in-service during plant operation and shutdown.

The operability of each of the seismic instrumentations is demonstrated by performing channel checks every two weeks for the first three months of service after startup. After the initial three-month period and three consecutive successful checks, the channel checks are performed on a monthly basis. The channel calibrations are performed every 24 months or during each refueling outage. The channel functional test is performed every 6 months. At least once a year, the system is operated continuously on the standby power source to verify the required backup power availability per CSA N289.5, Clause 9.2.2.

The guidance of Appendix A to U.S. NRC RG 1.166 (Reference 3.3-41) is followed for instrumentation found to be out of service during an earthquake.

3.3.1.5.4 Arrangements for Control Room Operator Notification

In accordance with the guidance of U.S. NRC RG 1.12, Section C.4.13, the triaxial accelerograph system is triggered whenever a threshold free-field acceleration of not more than 0.01 g is exceeded for any of the three axes. A higher threshold value can be used if 0.01 g is impracticable due to the site geological or geotechnical conditions or the ambient noise at instrument locations.

Activation of the seismic trigger causes an audible and visual annunciation in the control rooms to alert the plant operator that a felt earthquake has occurred in accordance with Clause 5.1.3 of CSA N289.5. Authorities having jurisdiction as well as the local and regional emergency response agencies are advised of the plant status if an earthquake exceeds the threshold acceleration per Clause 6.5.4 of CSA N289.1.

3.3.1.5.5 Comparison of Measured and Predicted Responses

The appropriate response after a felt seismic event is determined by the level of shaking. In accordance with Clause 6.5.1 of CSA N289.1, the BWRX-300 post-seismic plant operation manual defines the response associated with each level of shaking. The required operator actions after a felt earthquake are in accordance with Clause 6.5.7 of CSA N289.1.

Per Clause 6.5.5 of CSA N289.5, an immediate shutdown of the plant is not mandatory if during and following an earthquake the plant continues successful operation. The plant is shut down if it is determined that the earthquake intensity exceeded the DBE or if there is evidence of damage impacting the safety systems.

In the event of a plant trip, all records pertaining to fuel and reactor internals systems are compared to the data that are recorded during a normal shutdown and/or previous plant trips. The intensity of the earthquake and any evidence of damage will dictate if a detailed inspection is required or if a restart is allowed. Prior to startup, the availability of all safety class SSC is confirmed to ensure they can perform their intended functions.

Immediate Response Following a Seismic Event

If the plant remains online following a seismic event, the immediate response is to stabilize the plant in accordance with Clause 6.5.7.1.1 of CSA N289.1 by:

- Testing all systems required to perform nuclear safety functions
- Initiating inspections performed in accordance with the provisions of ANSI/ANS-2.23 (Reference 3.3-42) to assess the intensity of the seismic events and the effects on essential systems

Recorded earthquake data from the seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination of whether the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event.

Seismic Design Basis Exceedance

Following a seismic event, records of free-field ground motion and in-structure responses are reviewed in accordance with Clause 6.5.6.1 of CSA N289.1.

Cumulative absolute velocity calculated in accordance with Section 6.4.1 of ANSI/ANS-2.23 and peak ground velocity are generated from all free-field ground motion to be used as damage indicators. Damage criteria for Heavy industrial SSC in Section 6.5.6.2.1 of CSA N289.1 are also considered to help determine seismic design basis exceedance.

The DBE is considered exceeded when the measured free-field motion in any of the three directions (two horizontal and one vertical) exceeds the following limits:

1. Response spectrum limit that is exceeded if:
    a. At frequencies between 2 and 10 Hz, the recorded response spectral accelerations of 5% damping exceed the corresponding DBE design acceleration response spectrum or 0.2 g, whichever is greater or
    b. At frequencies between 1 and 2 Hz, the recorded response spectral velocities of 5% damping exceed DBE velocity response spectrum or 152 mm/sec, whichever is greater
2. Cumulative absolute velocity limit that is exceeded if the cumulative absolute velocity value calculated in accordance with Clause 6.5.6.1 of CSA N289.1 is greater than 0.16 g-s, or the peak ground velocity is greater than 50 mm/s.

The DBE exceedance is checked for measurements taken from the free-field plant grade accelerometers and downhole accelerometers using the corresponding design response spectra defining the DBE ground motion at the plant grade and RB foundation bottom elevations.

In addition to the criteria above, the following is also used to determine DBE exceedance:

- The inspection of the seismically qualified SSC shows evidence of overstressing, large displacement, yielded supports, etc.
- If the data collected from the monitoring instruments installed at different elevations in the plant exceed the DBE response parameters at the corresponding locations

Required Pre-Shutdown Earthquake Actions

Prior to the shutdown, the availability of safety class systems required for shutdown and the availability and integrity of the containment system are confirmed by performing pre-shutdown checks in accordance with the provisions of CSA N289.1, Clause 6.5.7.2.

Post-Shutdown Earthquake Response Actions

While the plant is shut down, a detailed inspection and evaluations are performed to assess the state of the plant in accordance with the provisions of CSA N289.1, Clause 6.5.7.3.

Post-shutdown actions include:

- Focused inspections of a preselected set of SSC that are representative of a broad cross section of equipment and structures in nuclear and conventional power plants
- Expanded inspections if damage is found in focused inspections
- Further graded inspections, tests, and analyses that are guided by the damage and earthquake levels

Focused inspections include detailed, visual inspections and tests of a preselected sample of representative structures and equipment, selected to sample all types of safety class and SCN SSC that are considered most likely to be damaged due to earthquake shaking. SCN SSC that experience has shown to be of low seismic capacity to serve as earthquake damage indicators are also included in the focused inspections.

Expanded inspections and tests are performed if significant physical or functional damage is found during the focused post-shutdown inspections. The expanded inspections include all accessible safety class equipment and structures as well as non-safety-class balance-of plant equipment that is important to safe operation of the plant. Expended 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.

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.

The structural integrity of the CB is maintained in the event of a design basis tornado missile to allow egress of operators to the Secondary Control Room (SCR) in the RB and to ensure availability of SSC providing post-disaster mitigation functions. For the special hardening provisions considered in the design of the CB, refer to Chapter 9B, Section 9B.3.2.2.

For a discussion of tornado dampers used to protect the Heating, Ventilation and Air Conditioning (HVAC) openings in the RB and CB to improve their survivability under tornado, refer to Chapter 9A, Section 9A.5.

The procedures for transforming tornado wind speed into pressure-induced forces to apply to structures and the distribution across the structures are based on BC-TOP-3-A (Reference 3.3-46). U.S. NRC RG 1.76 provides guidance to determine the pressure drop and rate of pressure drop caused by the passage of a tornado.

Missiles created as a result of components and cladding failing during a tornado wind event are considered enveloped by the design basis missile spectrum considered for the RB.

3.3.2.6 Hurricanes

Hurricanes at the DNNP site are considered bounded by tornado loads discussed in Subsection 3.3.2.5.

3.3.2.7 Lightning

Complying with Section 7.4.2 of CNSC REGDOC-2.5.2, grounding and lightning protection systems are used to protect structures, transformers and equipment against lightning induced surges as described in Chapter 8, Section 8.6.

Protection measures against fires and electromagnetic compatibility issues that could affect the functionality of electrical systems as a result of lightning are addressed in Subsections 3.3.6 and 3.3.7.1.

3.3.2.8 Extreme Wind Interaction

As described in Subsection 3.3.2.5, evaluations are performed to ensure that there is no adverse interaction between the RWB, CB, TB and Reactor Auxiliary Bay and the RB under design basis tornado wind loads applicable for the RB.

3.3.3 Extreme Hydrological Conditions

Potential sources of external floods considered in the BWRX-300 design are discussed in Chapter 2, Subsection 2.5.3.

To conform with Section 7.4.2 of CNSC REGDOC-2.5.2 and in accordance with U.S. NRC RG 1.102, Seismic Category A and RW-IIa structures are designed to include protective features that are used to mitigate or eliminate the adverse consequences of flooding due to external sources.

Conforming with CNSC REGDOC-2.5.2, Section 7.15.1, the integrated RB structure is designed to withstand the maximum external flood and groundwater levels specified in Chapter 2, Section 2.5.3.1.

Protection measures considered for the integrated RB structure against underground water includes the use of:

1. Hydrostatic and hydrodynamic loads to design walls below flood level in conformance with CNSC REGDOC-2.5.2, Section 7.15.1
2. Suitable provisions to ensure water tightness of external surfaces and penetrations below design basis maximum flood and groundwater levels

3. No exterior access openings below grade

In accordance with U.S. NRC RG 1.143, the RWB is designed for one-half of the Probable Maximum Flood (PMF) listed in Chapter 2, Subsection 2.5.3.1.

Because plant grade is above design flood level, the Power Block structures remain accessible during postulated flood events. Thus, no emergency actions are required due to flooding to ensure the safe operation of the BWRX-300 plant.

3.3.3.1 Analysis Procedure

The BWRX-300 RB is analyzed and designed to withstand the effects of the maximum external flood and highest groundwater levels specified for the plant. The maximum flood and highest groundwater levels listed in Chapter 2, Subsection 2.5.3.1 are considered in defining the input design parameters for the structural design to account for flood and groundwater loadings.

Because the flood level at the DNNP site is below the finished grade level, only hydrostatic effects are considered in the analysis and design of structures, while dynamic phenomena associated with a flooding event, such as currents, wind waves, and their hydrodynamic effects are not considered. The hydrostatic pressure associated with the design flood level or with the design groundwater level is considered as a structural load on the basemat and basement walls for structural design in accordance with CNSC REGDOC-2.5.2, Sections 7.4.2 and 7.15.1. Uplift or floating of structures is considered and the total buoyancy force is based on the hydrostatic pressure due to the design flood level, excluding wave action, or the design groundwater level. The lateral, overturning and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, are also considered in the structural design of these elements.

3.3.4 Aircraft Crash

This section discusses non-malevolent, general aviation crashes in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.4.2. For robustness against malevolent acts, including aircraft crashes, refer to Subsection 3.3.7.4.

Small aircraft crashes are considered in the BWRX-300 design but are screened out per Chapter 2, Subsection 2.2.3.1. The design considers these aircraft crashes as missiles bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

To mitigate their potential of equipment damage and fire impacts, the design of the BWRX-300 Seismic Category A structures addresses penetration resistance of buildings and considers physical separation of redundant or backup equipment, where applicable.

3.3.5 Missiles

3.3.5.1 Missiles Generated by Extreme Winds

Refer to Subsection 3.3.2.5 for details.

3.3.5.2 Site Proximity Missiles (Except Aircraft)

The design considers site proximity missiles to be bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

Due to the distance between the sites, the maximum turbine missile from the existing Darlington site does not impact the DNNP site.

3.3.5.3 Structures, Systems and Components to be Protected from Externally Generated Missiles

Seismic Category A, RW-IIa, and portions of the TB and CB structures are designed to withstand the effects of externally generated missiles. For Seismic Category A SSC, the tornado wind characteristics and tornado missile spectra considered in the design are listed in Chapter 2, Table 2.6-3 and Table 2.6-4. Tornado wind and tornado missile spectra design input values considered in the design of the RWB are listed in Table 2 of RG 1.143.

The response determination methodology due to missile impact loading on the RB structure, consisting of Steel Bricks™ modules, is in accordance with ANSI/AISC N690, Appendix N9.1, Section 6c.

The response determination methodology due to missile impact loading on the RWB and portions of the TB and CB is in accordance with CSA N291, Annex A.

3.3.5.4 Barrier Design Procedures

In accordance with CSA N291, Clause A.5, barrier design for impact loads satisfies the criteria for local and overall effect. The procedures for designing barriers to withstand the effects of missile impacts are per U.S. NUREG-0800, SRP 3.5.3.

3.3.5.4.1 Local Damage Prediction

The prediction for local damage in the impact area depends on the basic material of construction of the barrier.

Concrete Barriers

Sufficient thickness of concrete is provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact.

Per CSA N291, Clause A.5.2.3, empirical formulas are applicable over a limited range of missile and target parameters.

Required concrete barrier thicknesses are determined in accordance with U.S. NUREG-0800, SRP 3.5.3 and are in no case less than those of Region I listed in Table 1 of U.S. NUREG-0800, SRP 3.5.3. In accordance with CSA N291, Clause A.5.2.4, the required barrier or wall thickness to prevent perforation is at least 20% greater than the calculated thickness from the applicable empirical formulas. Also, the required barrier or wall thickness to mitigate missile penetration is at least 50% greater than the calculated thickness from the applicable empirical formula.

Steel Barriers

Steel barrier thicknesses are determined using the Stanford equation (Reference 3.3-48) in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

Composite Sections

Composite section barriers are utilized in the BWRX-300 for missile protection when the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element for prediction of local damage in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

3.3.5.4.2 Overall Damage Prediction

The BWRX-300 design for impactive loads satisfies the criteria for the overall effect of Clause A.5.3 of CSA N291. Dynamic effects of impactive loads are evaluated by dynamic analysis in accordance with Clause A.4.1.1 of CSA N291 or the equivalent static load approach mentioned in Clause A.4.1.2 of CSA N291.

3.3.5.4.3 External Doors

The RB external doors are designed to resist tornado missiles unless shielded by external stair towers or elevator shafts. External stair towers or elevator shafts credited for shielding are evaluated for tornado missiles.

3.3.6 External Fires, Explosions and Toxic Gases

In line with requirements of CNSC REGDOC-2.5.2, Section 7.4.2, damages due to fires, explosions, and release of toxic gases as a result of transportation and industrial accidents at or near the DNNP site are considered in the BWRX-300 design. The following subsections provide information on measures considered to protect and mitigate the effects of:

- External fires – Subsection 3.3.6.1
- Explosions – Subsection 3.3.6.2
- Release of toxic gases – Subsection 3.3.6.3

3.3.6.1 External Fires

Per Chapter 2, Subsections 2.2.3, 2.2.4, 2.4.1 and 2.6.10, sources of external fires at the DNNP site include fireballs as a result of a rail transportation accident, forest fires, lightning and accidental fires in on-site storage areas of hydrogen, liquid waste or fuel oil. As mentioned in Chapter 2, Subsection 2.2.4, the risk of fire due to pipeline ruptures close to the DNNP site is negligible and is therefore not considered in the design.

Chapter 9A, Section 9A.6 describes the BWRX-300 fire protection systems implemented to resist and mitigate the effects of external fires. Buildings and structures within the protected area are supplied fire water from redundant loops by two fire water storage tanks (See Chapter 9A, Section 9A.6.6) providing suction to fire pumps located in a Fire Pump Enclosure structure (See Chapter 9B, Section 9B.3.6).

Figure A1.4-1 in Appendix A of Chapter 1 shows the location of the fire water storage tanks and Fire Pump Enclosure at the DNNP site.

Protection measures against the release of toxic gases as a result of external fires are discussed in Subsection 3.3.6.3.

3.3.6.2 Explosions

The RB structure is designed to withstand impulsive and impactive loads as discussed in Subsection 3.5.5.4.

3.3.6.3 Release of Toxic Gases

On-site activities that could result in release of toxic gases that could impact the safe operation of the BWRX-300 DNNP are summarized in Chapter 2, Section 2.4. External sources of toxic gases and chemicals are discussed in Chapter 2, Section 2.2.

Mitigation measures considered in the design of MCR/SCR are referenced in Chapter 6, Section 6.4.

3.3.7 Other External Hazards

3.3.7.1 Electromagnetic Interference

Protection against electromagnetic interference caused by lightning, high-voltage transmission lines at DNGS and telecommunication towers (See Chapter 2, Subsection 2.2.9) is provided through the use of appropriate shielding and qualification of equipment.

Safety Class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

For a description of plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements, refer to Chapter 8, Section 8.6.

3.3.7.2 Biological Phenomena

In accordance with CNSC REGDOC-2.5.2, Section 7.4.2, the Pumphouse/forebay structure is designed to prevent clogging by algae and exceptional quantities of fish and to stop them from entering the cooling systems. Measures considered to mitigate the effects of such clogging include locating the intake tunnel and lakebed intake structure at an adequate depth in the lake and the installation of traveling water screens to prevent intake of biofouling material as described in Chapter 9B, Subsection 9B.3.5.

As shown in Chapter 1, Appendix A, Figure A1.4-1, the BWRX-300 protected area is fenced which, in turn, prevents entry of large animals into the plant.

Screens or equivalent engineered features are also provided to prevent blockage of outside air intakes by non-human biota.

3.3.7.3 Collisions of Floating Bodies and Frazil Ice with Water Intakes

To satisfy requirements in CNSC REGDOC-2.5.2, Section 7.4.2, the design of the intake structure includes measures to mitigate the potential risk of blockage by frazil ice accumulations and physical damages as a result of a marine accident.

Measures considered to preclude blockage by frazil ice include a proper design of the Circulating Water System (CWS) recirculation line to prevent the formation of frazil ice in the forebay. Refer to Chapter 10, Section 10.8 for information related to the CWS.

To prevent marine transportation accidents, a restricted zone is established around the BWRX-300 lakebed intake structure and discharge diffusers to stop commercial ships from approaching offshore structures as stated in Chapter 2, Subsection 2.2.3.4.

3.3.7.4 Robustness Against Malevolent Acts

The BWRX-300 design provides robust physical features for the protection against malevolent actions found in the Design Basis Threats (DBTs) and Beyond Design Basis Threats (BDBTs). This results in the following fundamental capabilities remaining available after malevolent actions intended to cause substantial radiological releases:

- Ability to shut down the reactor and maintain sub-criticality
- Ability to cool irradiated fuel, both in the core and in the fuel pool
- Ability to limit or prevent the release of radioactivity affecting public health and safety

The ultimate gauge of success of the above three key functions is the prevention of radioactive releases that impact the health and safety of the public.

The BWRX-300 development has included a security by design approach from the early stages of design that uses sound engineering principles to demonstrate that, within an acceptable margin of confidence, sufficient capabilities are available to perform the above functions over a wide range of threats. This approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter.

The following are examples of features that enhance protection against malevolent actions:

- 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

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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."

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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 (1)	Limit State (2)
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 (3)	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/SCN	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

Table 3.3-2: Base Case Rock Dynamic Properties

Bedrock Formation	Total Unit Weight (kN/m^3)	Base Case Shear Wave			Poisson's Ratio
		V_s (m/s)	$\sigma_{\mu\ ln}$	$\sigma_{\mu\ lnVs}$	
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

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	$\sigma_{\mu\ ln}$	$\sigma_{\mu\ lnVs}$	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

Table 3.3-4: Rock Layers Kappa Values

Case	Bedrock Kappa ($\kappa_{0,\ ref}$; sec)	Rock Layer Kappa (κ_{r}; sec)	Total Kappa at Top of Rock (sec)
Base Case	0.006	0.002	0.008
Lower Realization	0.006	0	0.006
Upper Realization	0.006	0.006	0.012

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

Material	Response Level 1	Response Level 2
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

Table 3.3-7: Seismic Damping Values for Primary Subsystems

Component	Level 1 Damping		Level 2 Damping	
	Horizontal	Vertical	Horizontal	Vertical
Reactor Vessel	2	2	4.0	4.0
Vessel Support Skirt	2	2	4.0	4.0
Shroud	2	2	4.0	4.0
Shroud Support Spring	2	2		
Shroud Head & Separator	2	2	4.0	4.0
Fuel	4	4	6.0	6.0
CRD Guide Tubes	1	1	2.0	2.0
CRD Housing	1	1	2.0	2.0
CRD Restraint Springs	-	2		
Stabilizer and Bellows	-	2		
Welded Steel			4.0	4.0
Bolted Steel			7.0	7.0

Table 3.3-8: Preliminary Dynamic Loading α, β – Damping

Loading	Shell Model	Total Model Damping at A & B Freq	A Freq (Hz)	B Freq (Hz)	α	β
LOCA	52	6%	10	60	6.527	.000257
		6%	1.8	12.7	1.2083	.0011637
	110	4%	10	60	4.3731	.0001655
		4%	1.8	12.7	.8121	.0007246

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

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

Table 3.3-10: Piping and Equipment Damping Values for All Other Non-Seismic Loadings

Structure or Component	When considered by itself and/or combined with other load and designated as normal, upset and emergency	When considered by itself and/or combined with other load and designated as faulted
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

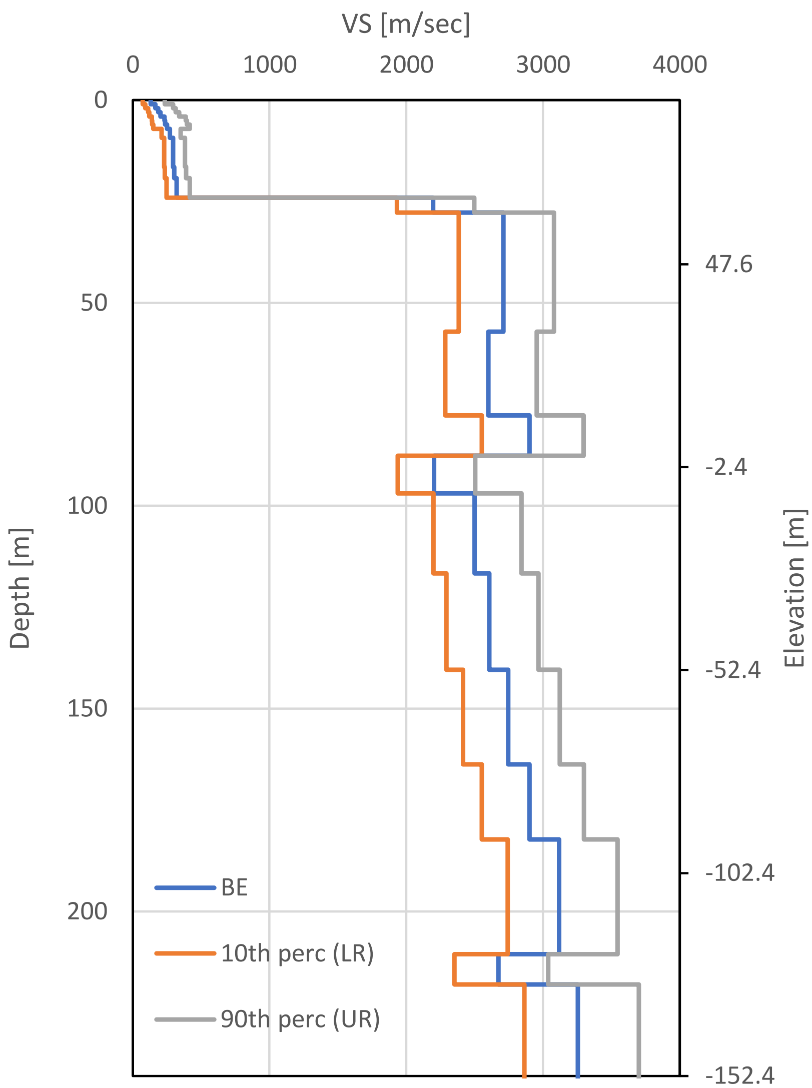


Figure 3.3-1: Shear Wave Velocities for the Bounding In-situ Profile

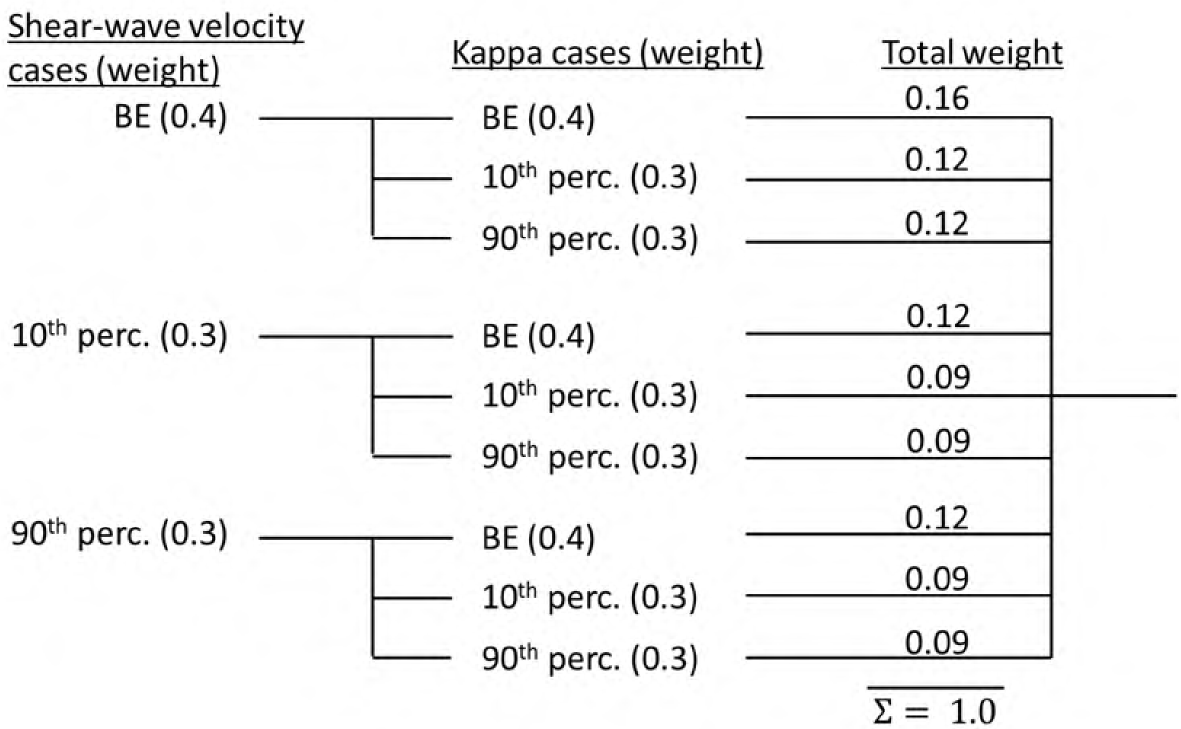


Figure 3.3-2: Cases Considered for Explicit Considerations of Epistemic Uncertainties

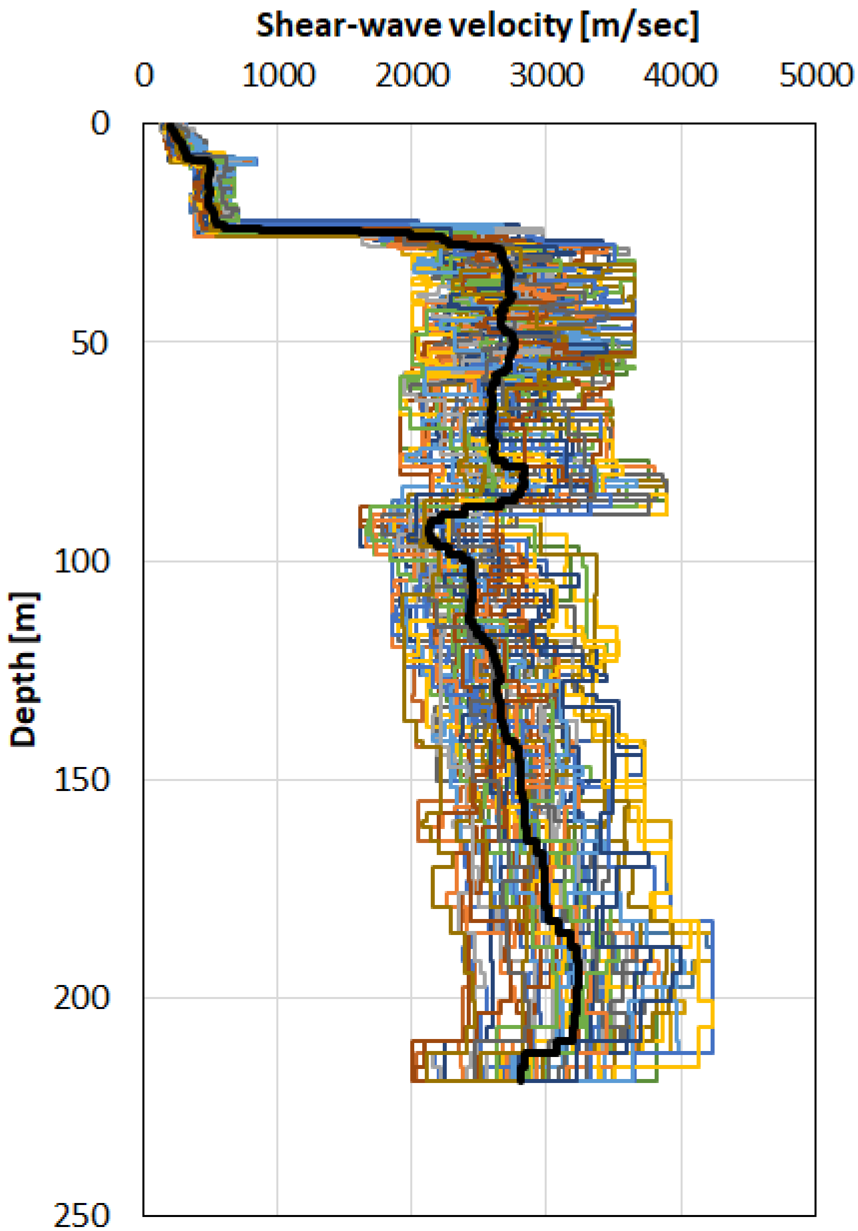


Figure 3.3-3: Shear Wave Velocity and Layer Thicknesses Randomization – BE-BE Case

Note: The Black line designates the resulting mean curve

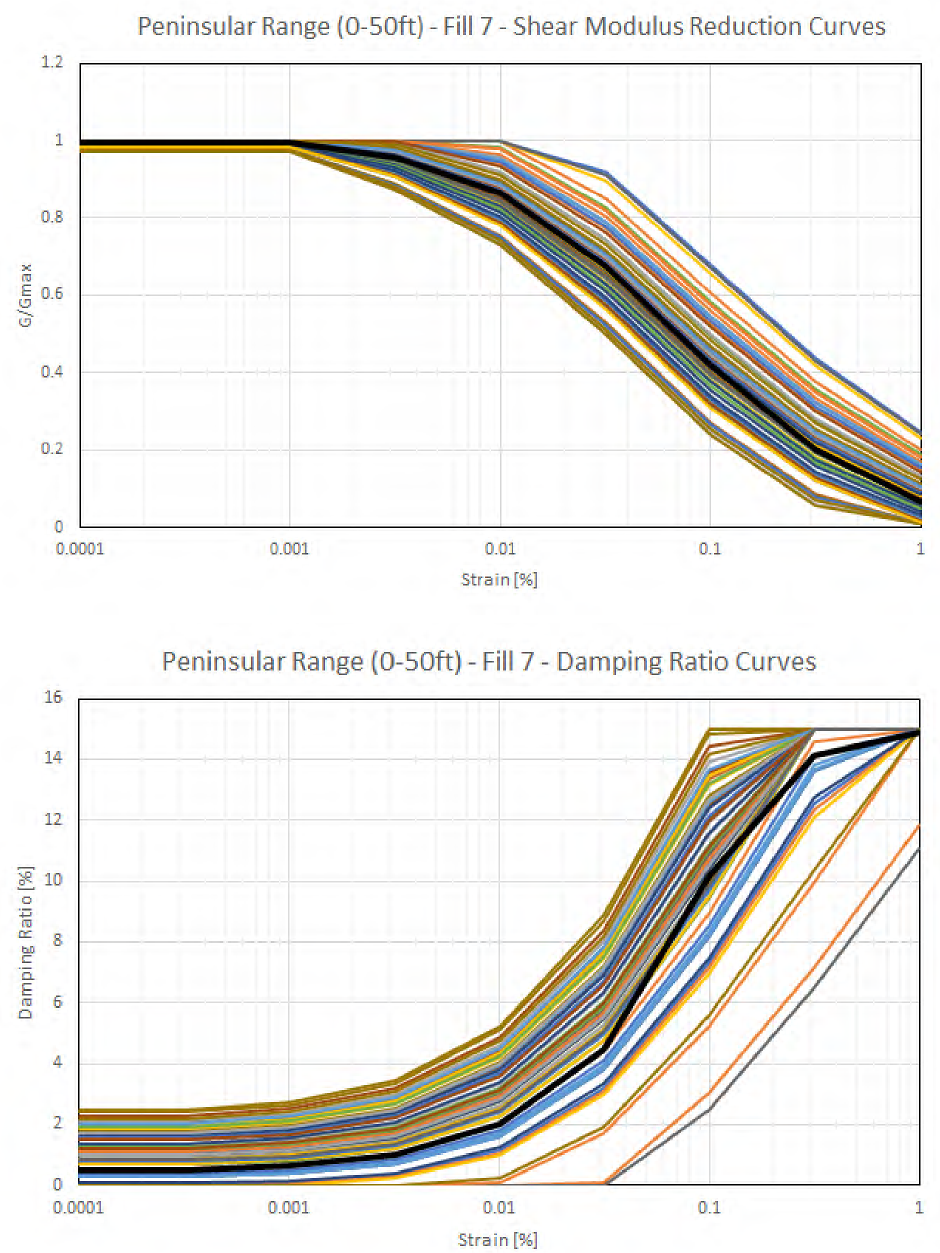


Figure 3.3-4: Soil Degradation Curves Randomization

Note: The Black line designates the resulting mean curve

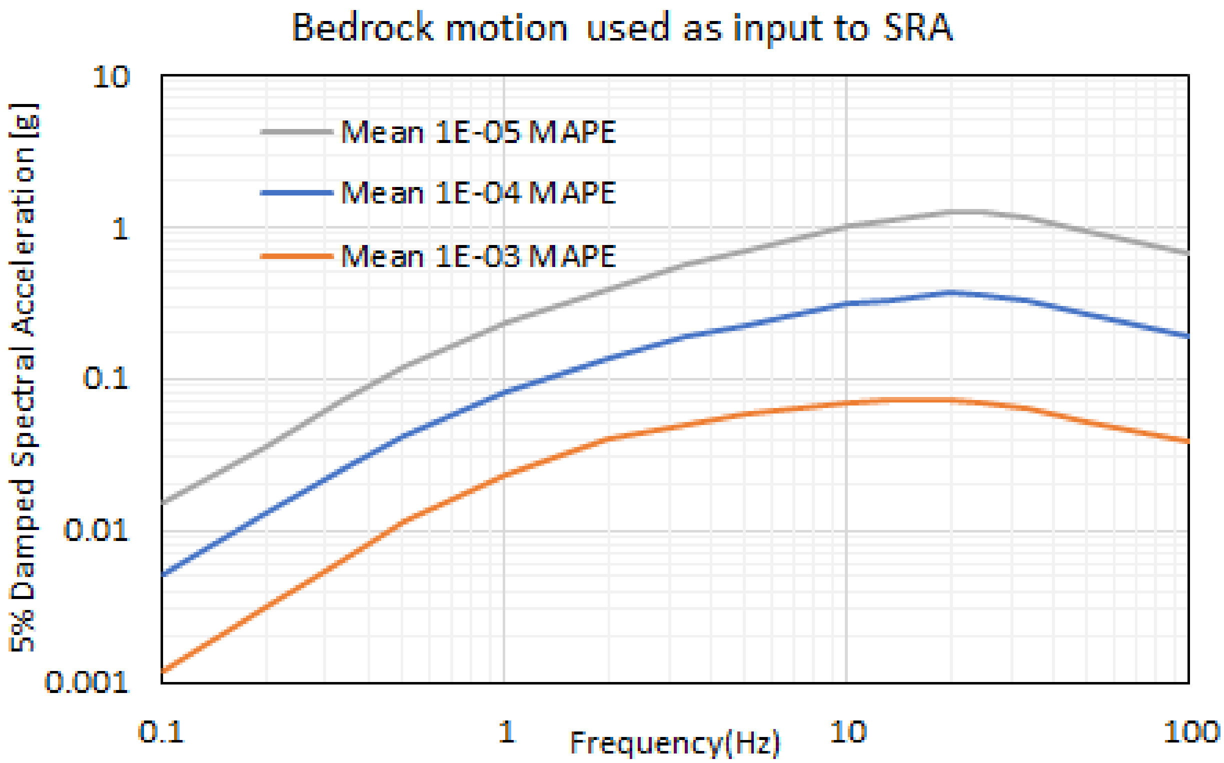


Figure 3.3-5: Uniform Hazard Response Spectra at Bedrock Elevation

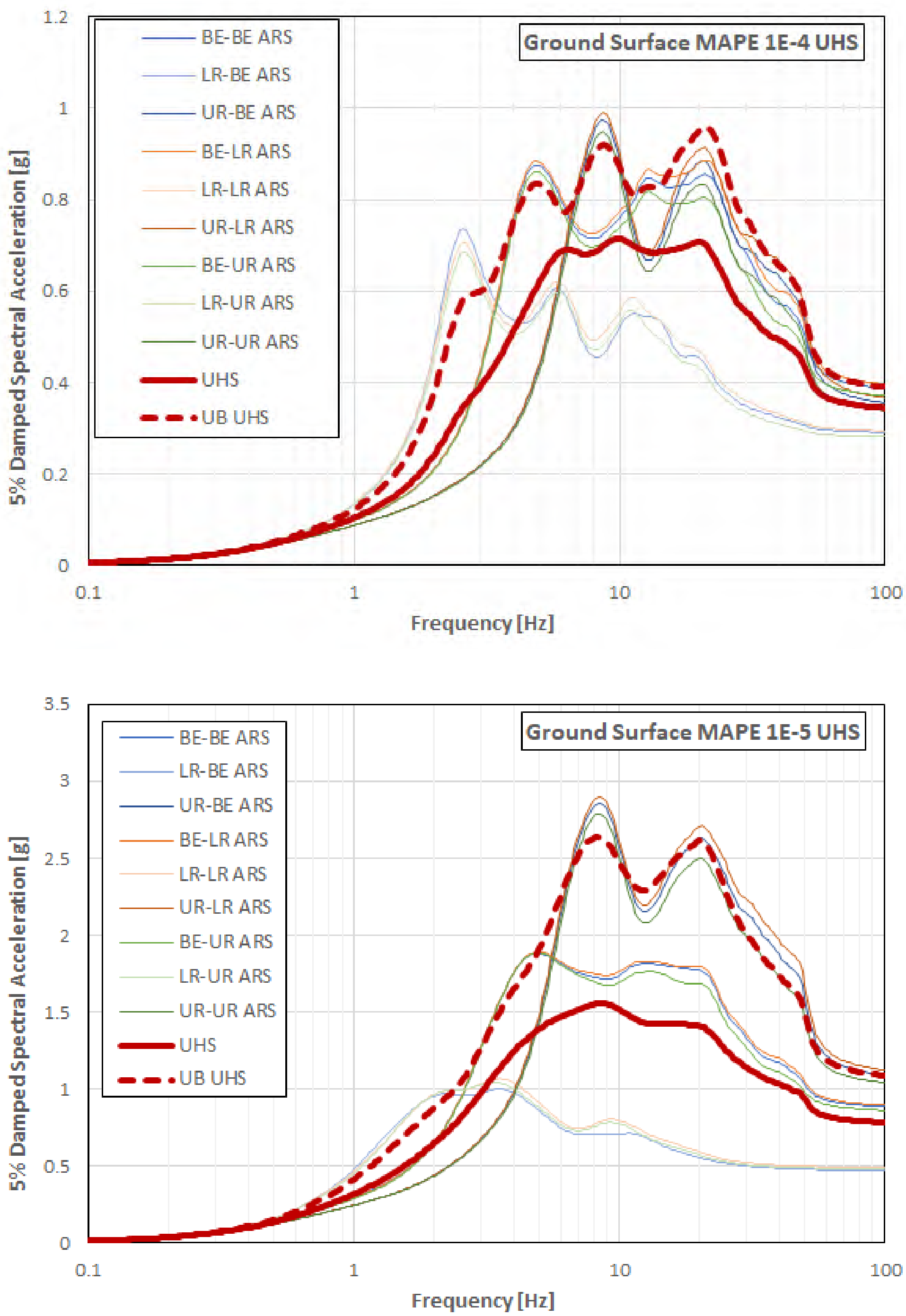


Figure 3.3-6: Ground Surface Uniform Hazard Response Spectra

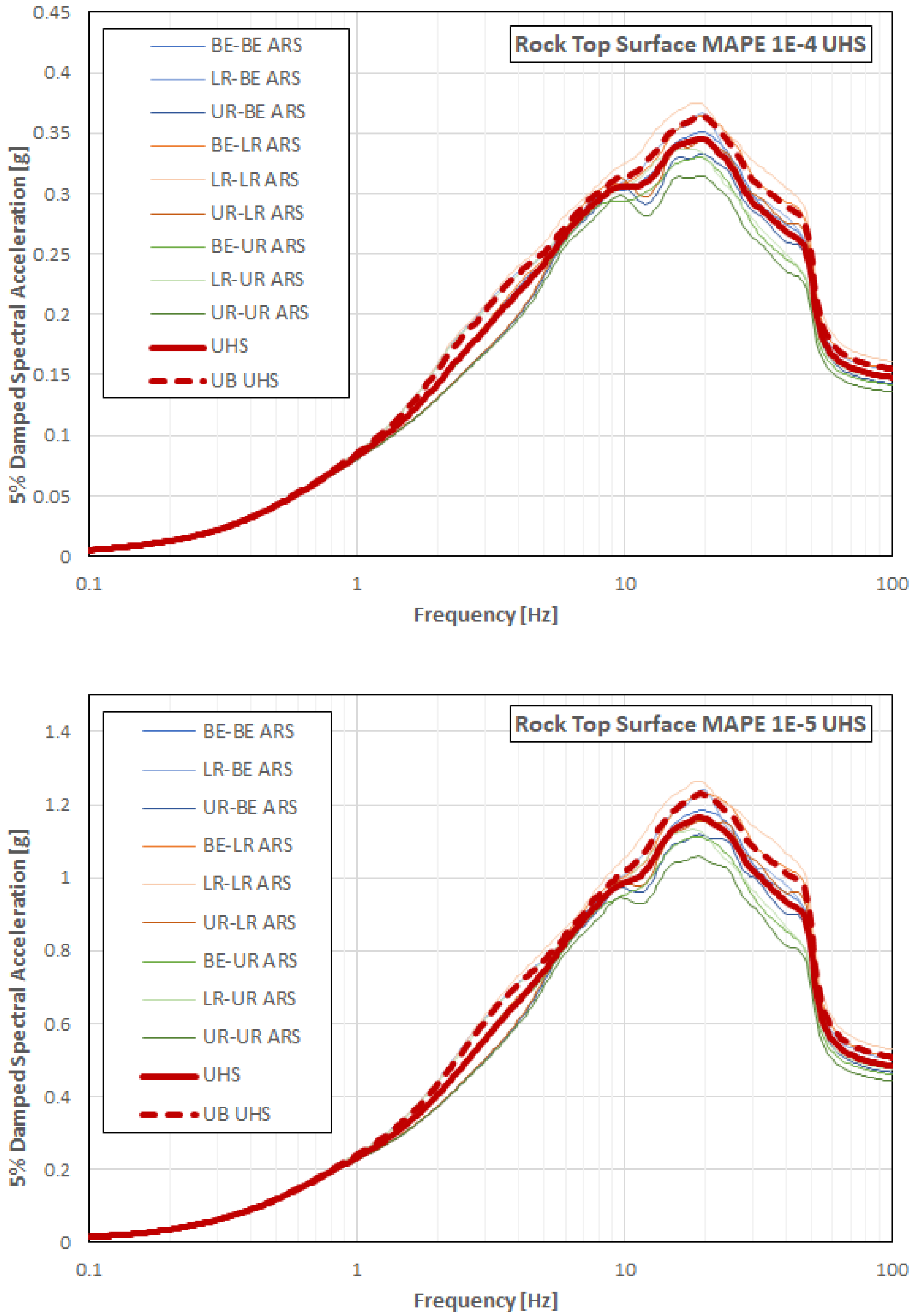


Figure 3.3-7: Rock Top Surface Uniform Hazard Response Spectra

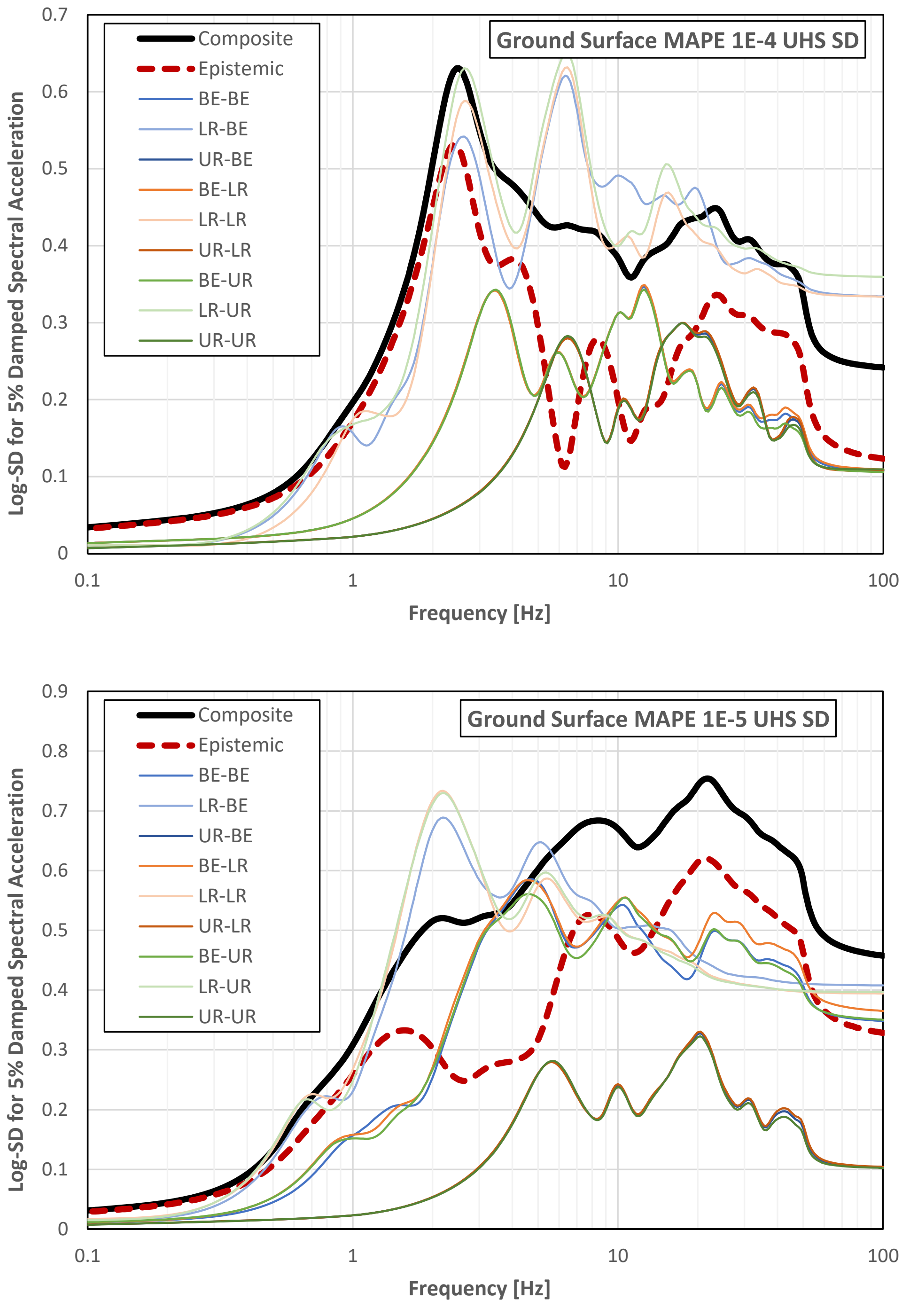
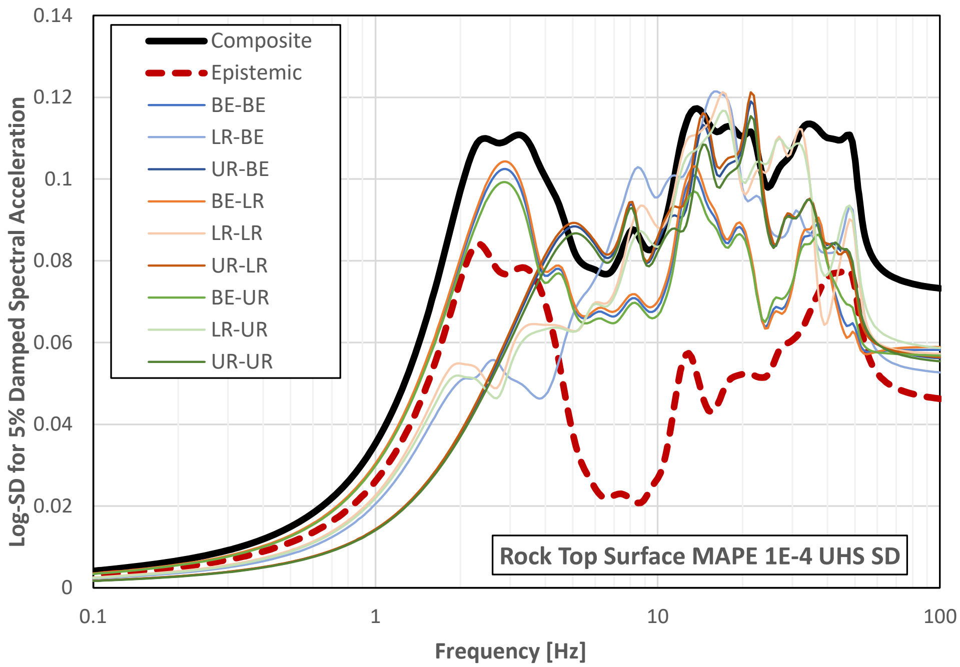


Figure 3.3-8: Ground Surface Composite and Epistemic Log-Normal Standard Deviations



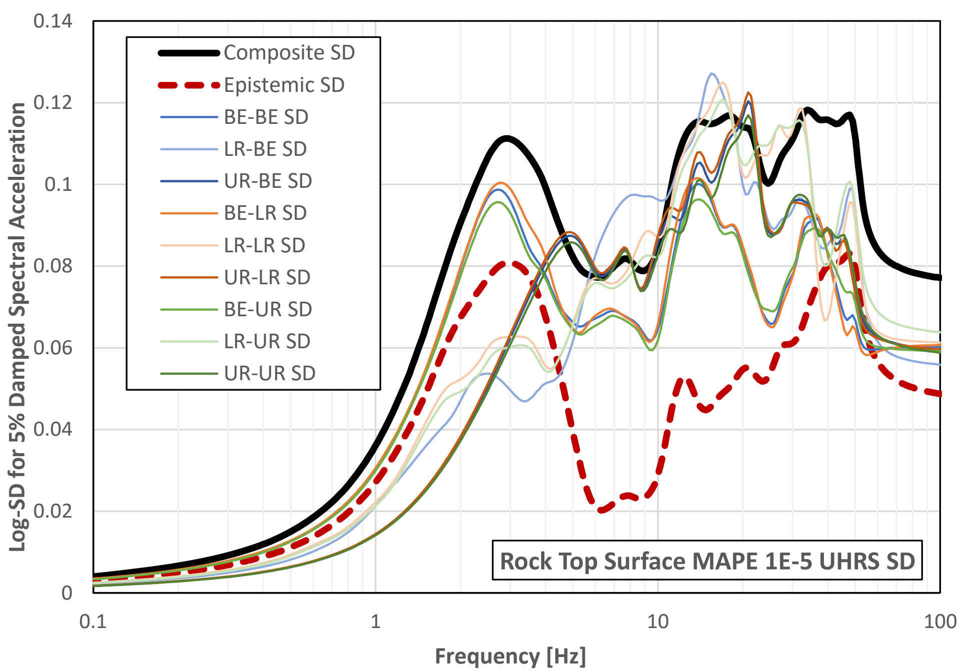


Figure 3.3-9: Rock Top Surface Composite and Epistemic Log-Normal Standard Deviations

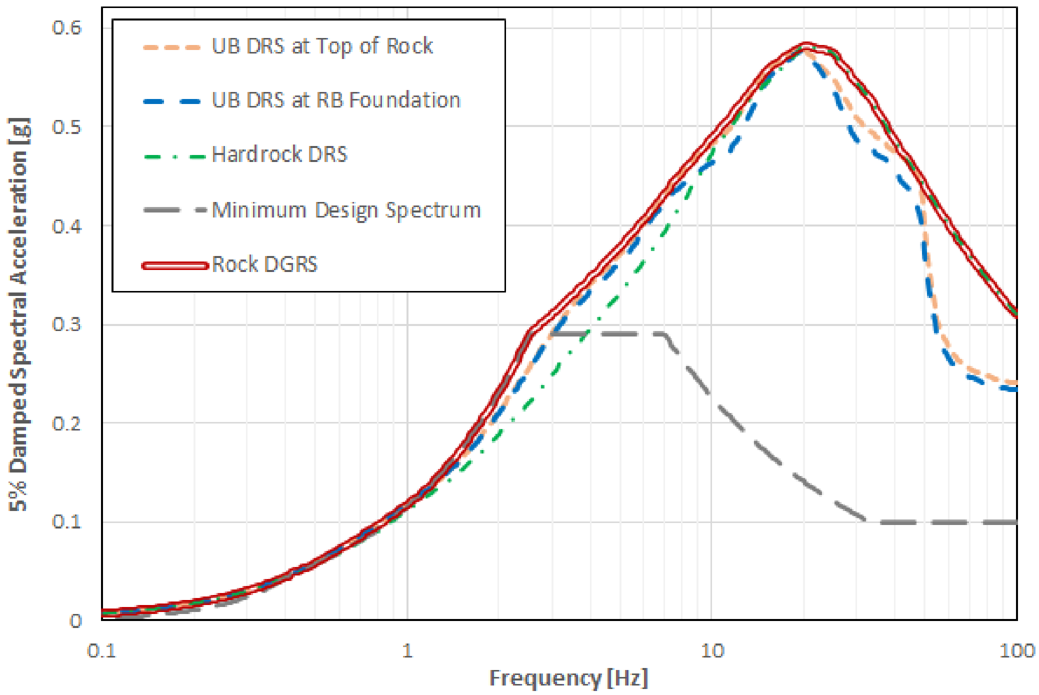
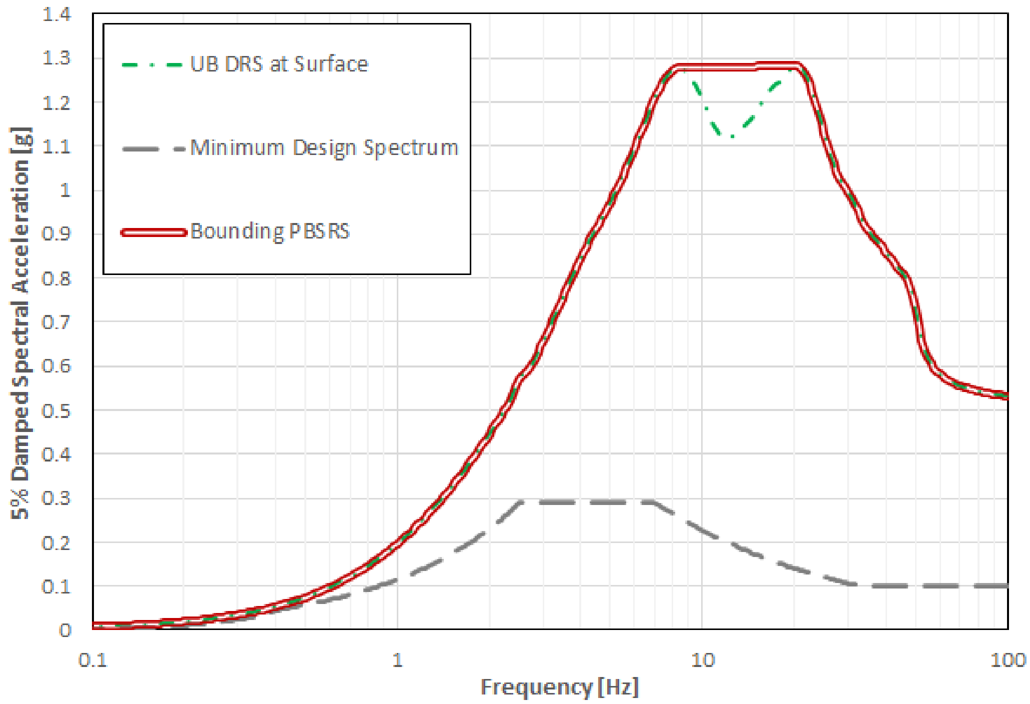


Figure 3.3-10: Bounding Horizontal Rock Ground Motion Response Spectra



nce Based

Surface Response Spectra

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

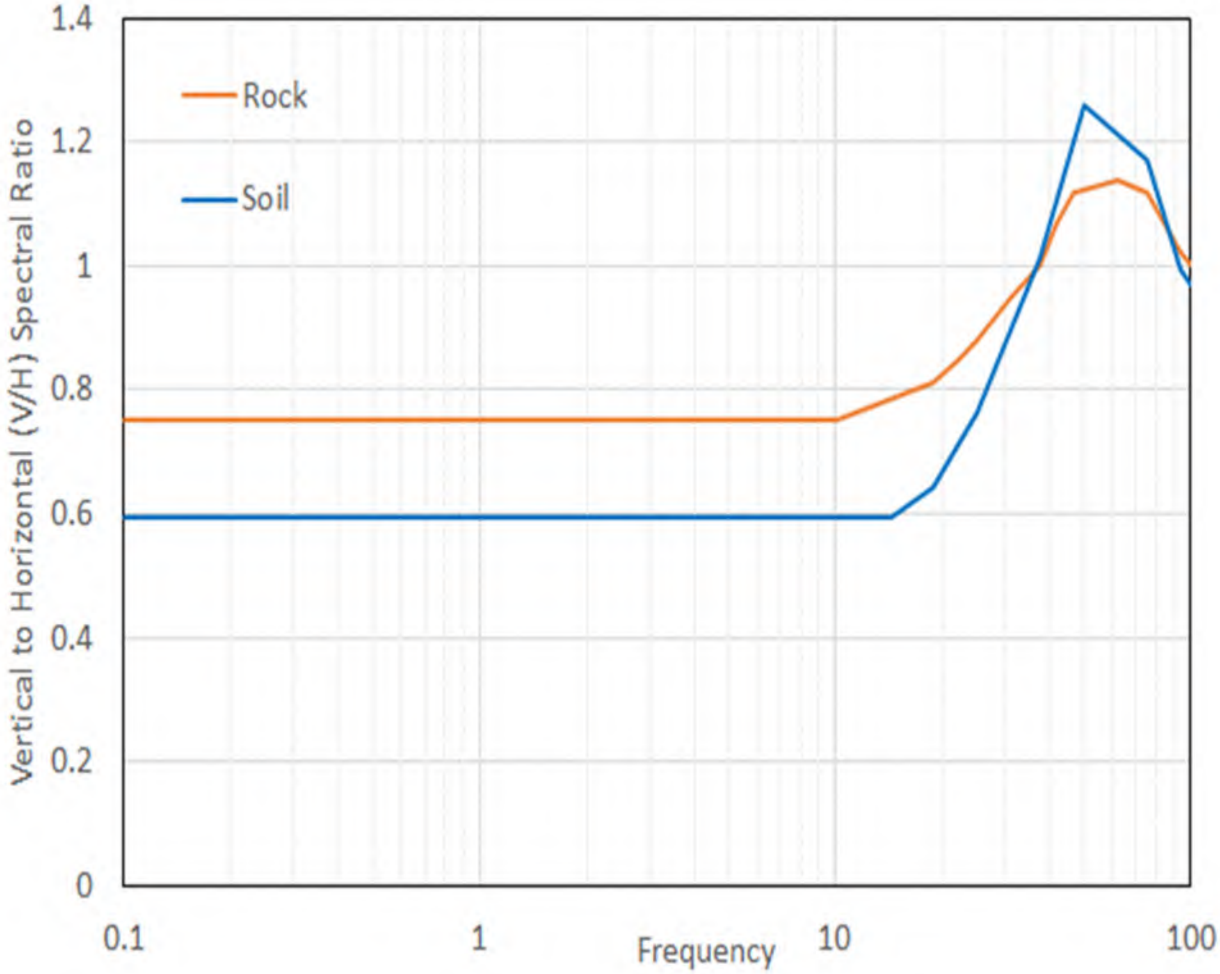


Figure 3.3-12: Vertical to Horizontal (V/H) Spectral Ratios

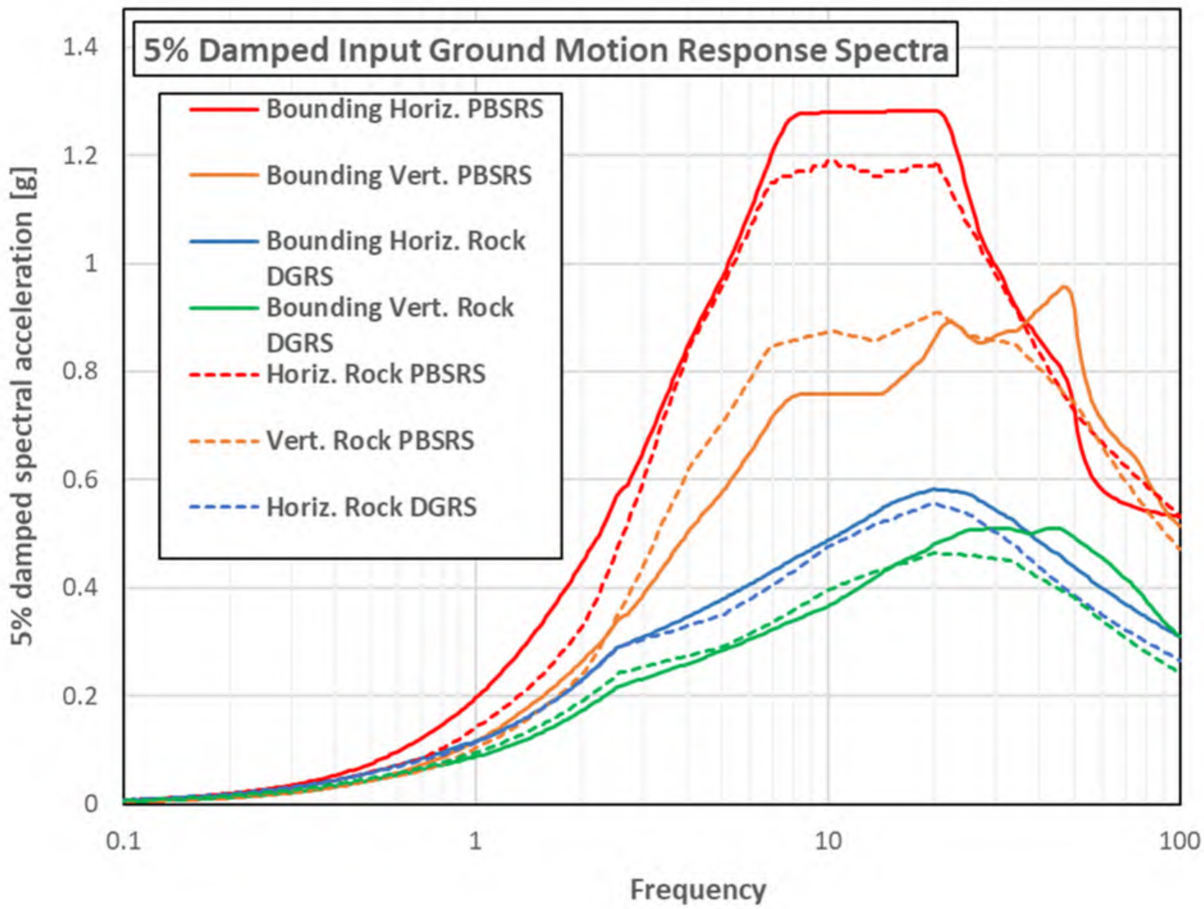


Figure 3.3-13: Comparison of Bounding to Updated Ground Motion Design Response Spectra

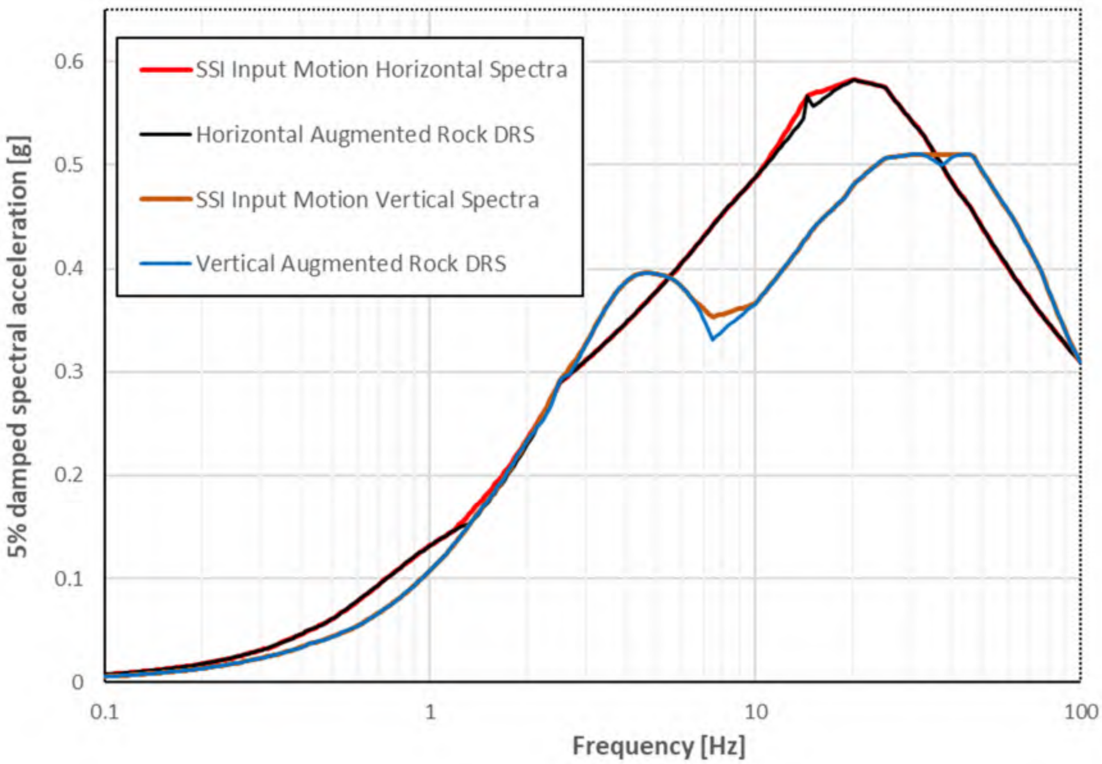


Figure 3.3-14: Augmented and Smoothed Horizontal and Vertical Rock Design Ground Response Spectra

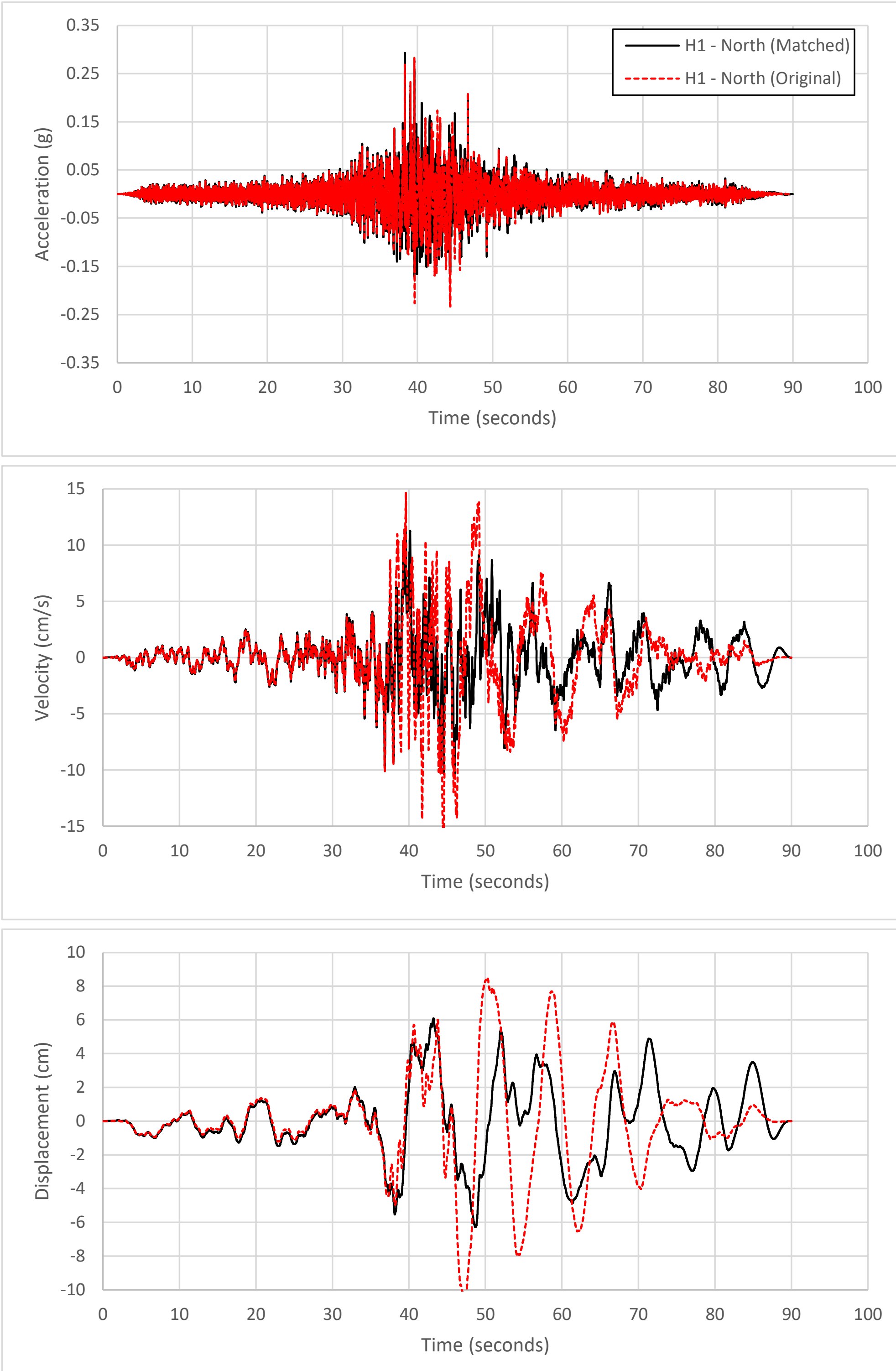


Figure 3.3-15: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 North

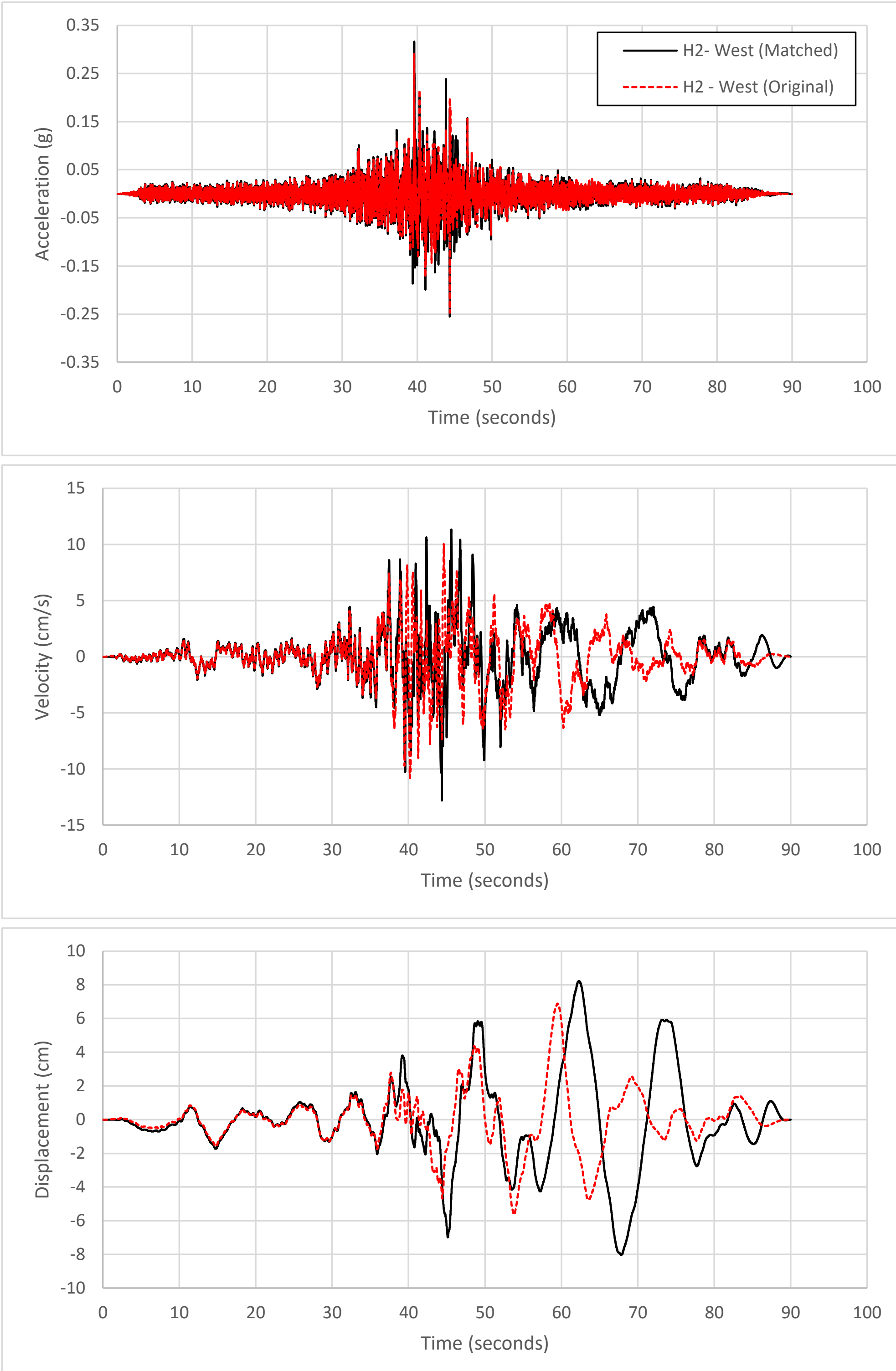


Figure 3.3-16: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 West

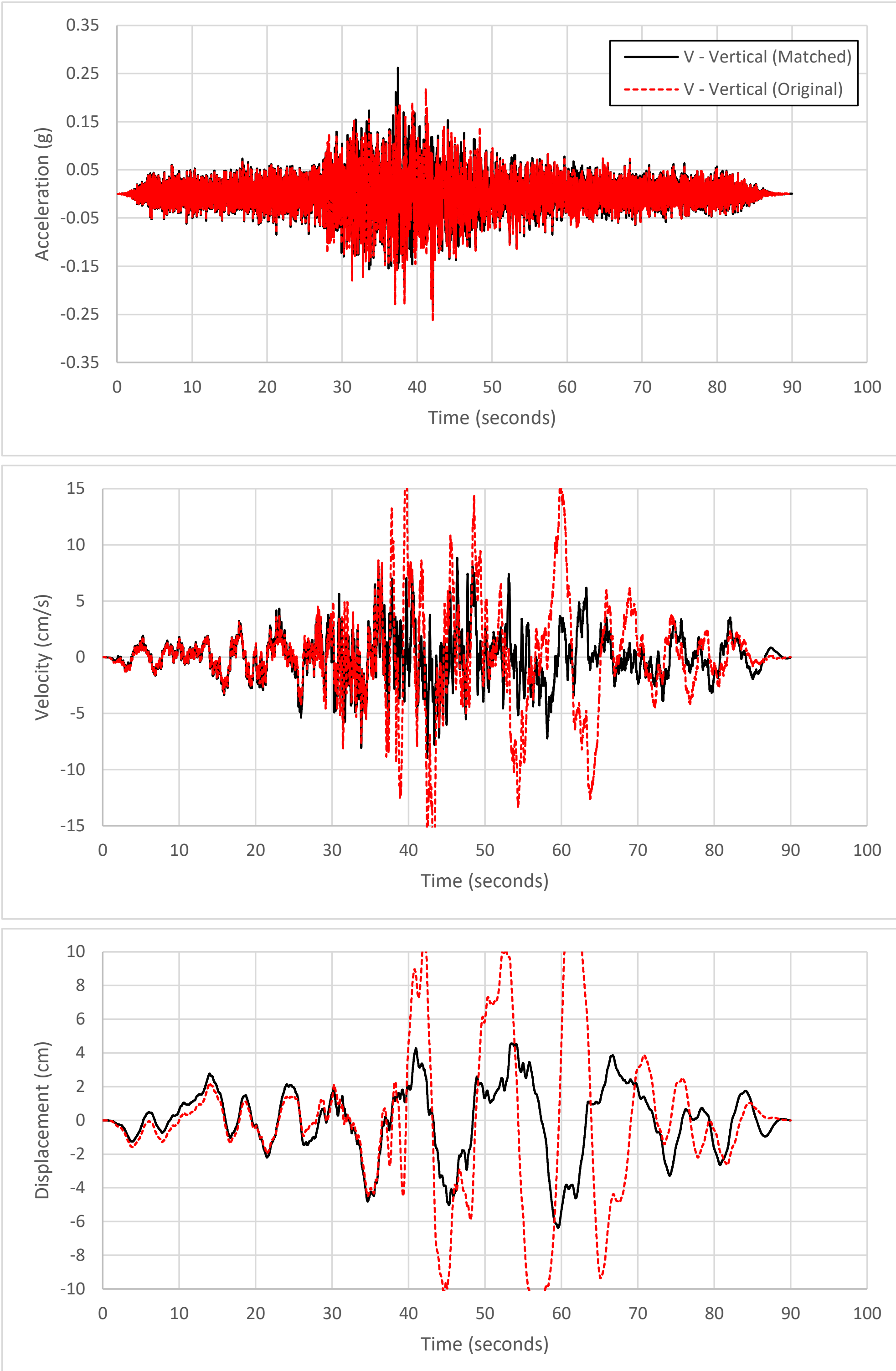
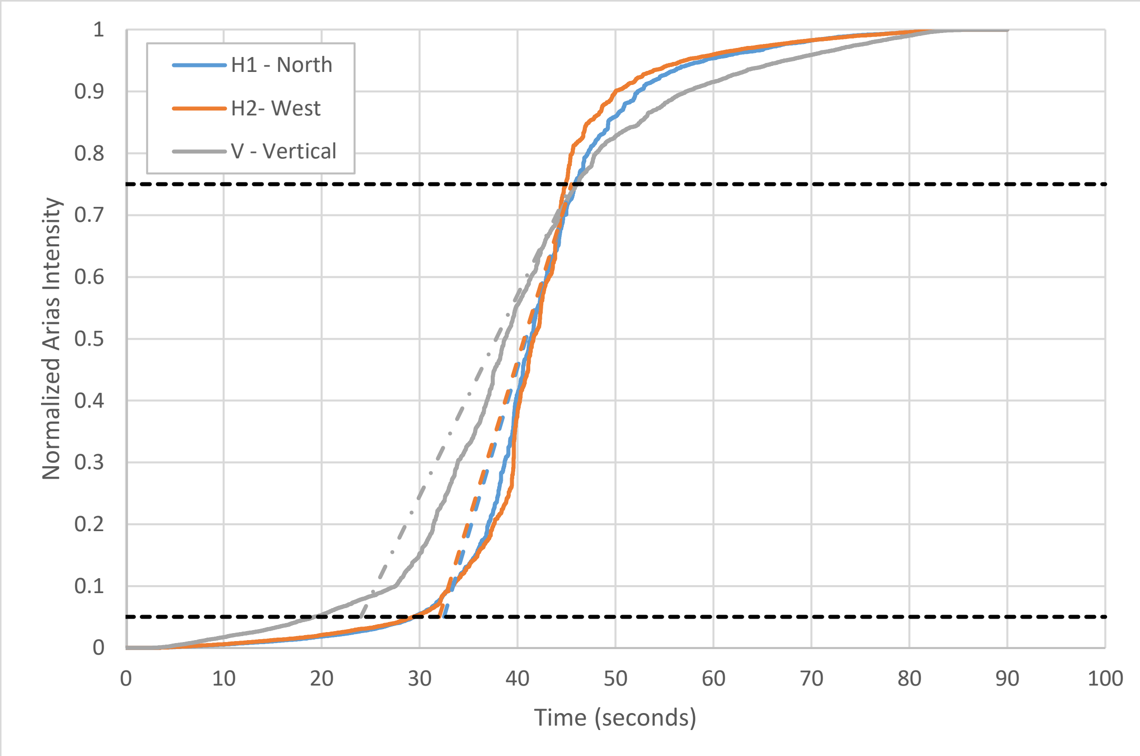


Figure 3.3-17: Vertical Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 Vertical



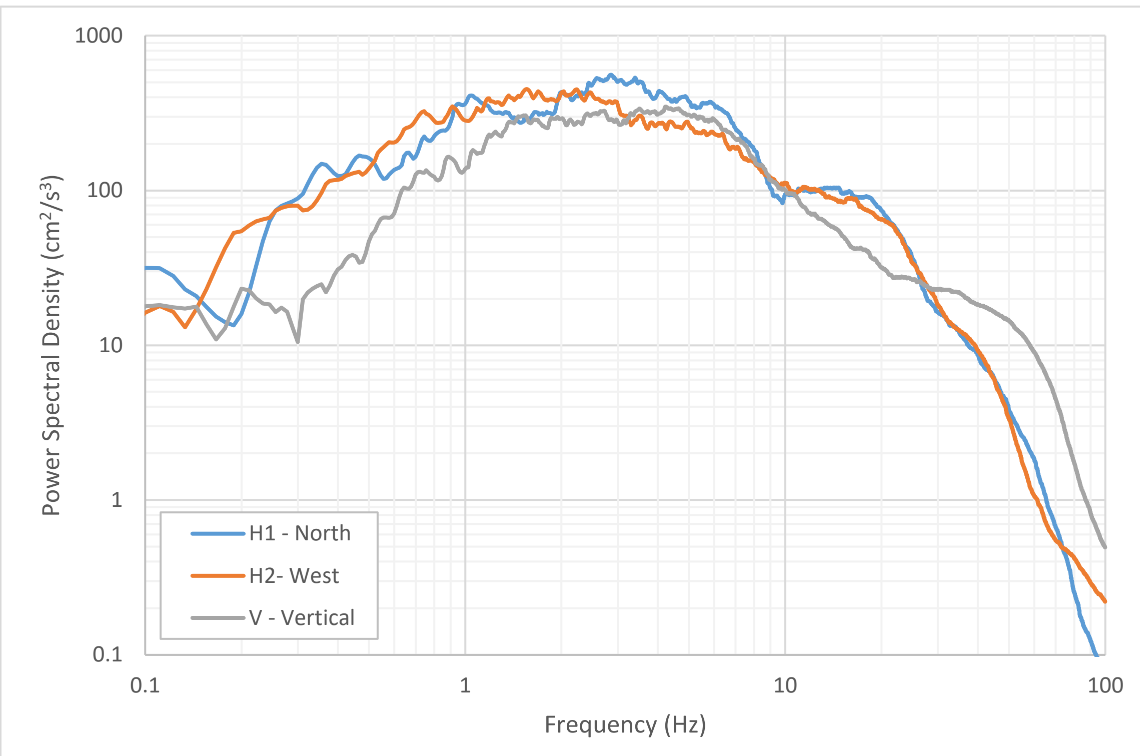
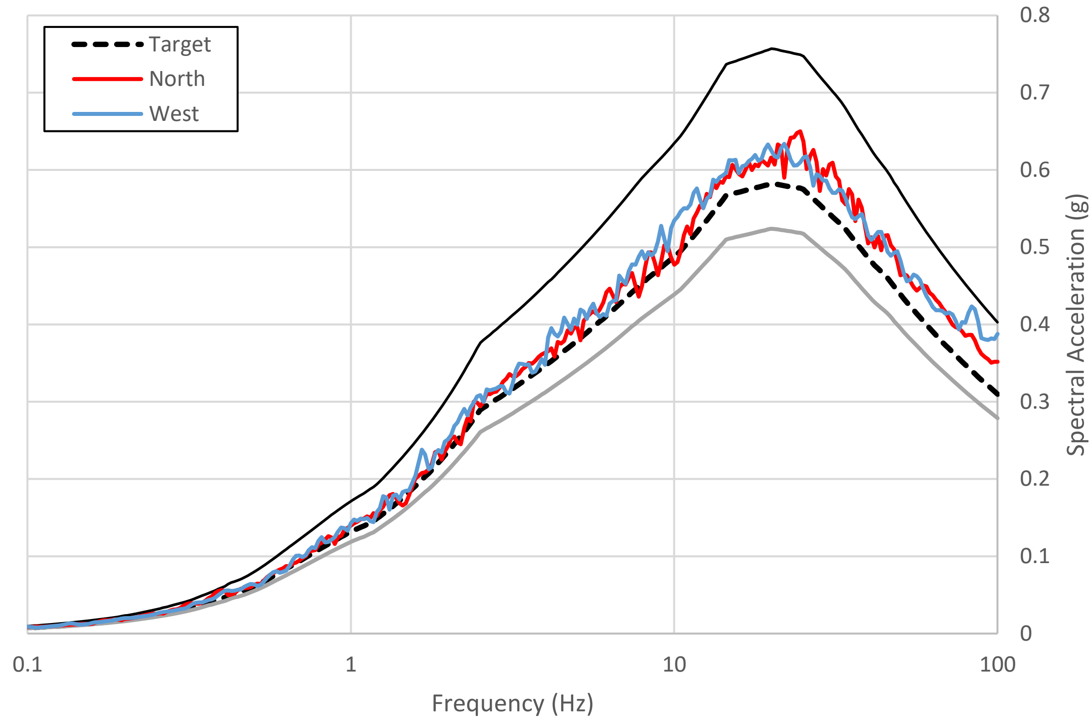


Figure 3.3-18: Normalized Arias Intensity and Power Spectral Density Function for Response Spectrum Matched HWA026 Acceleration Time Histories



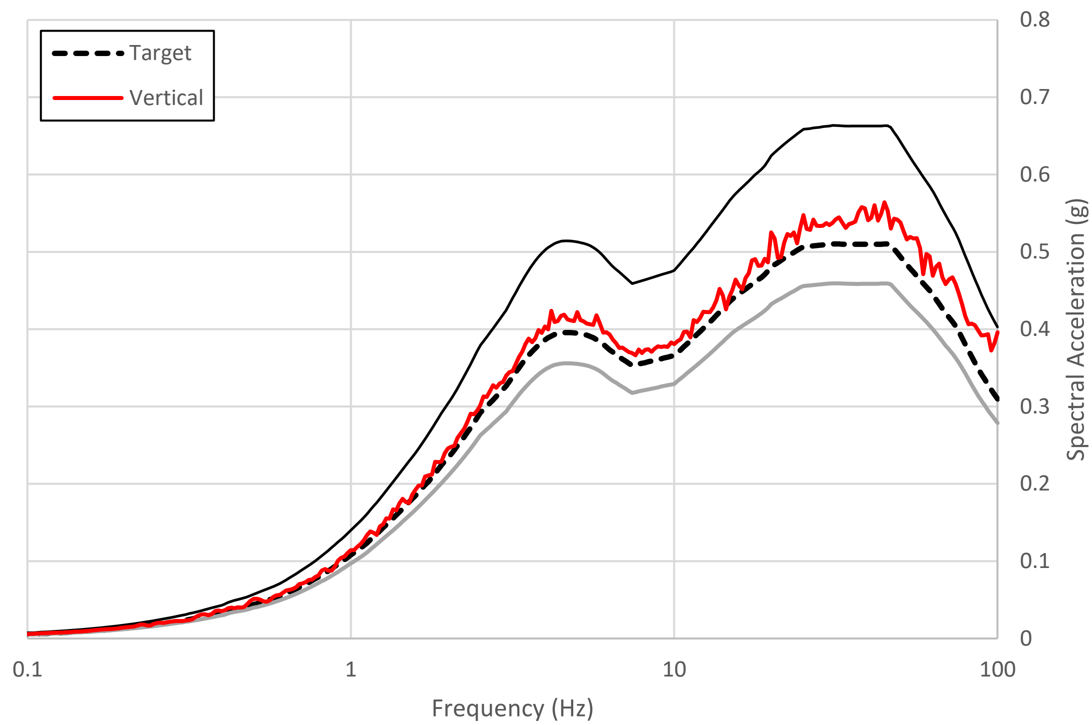
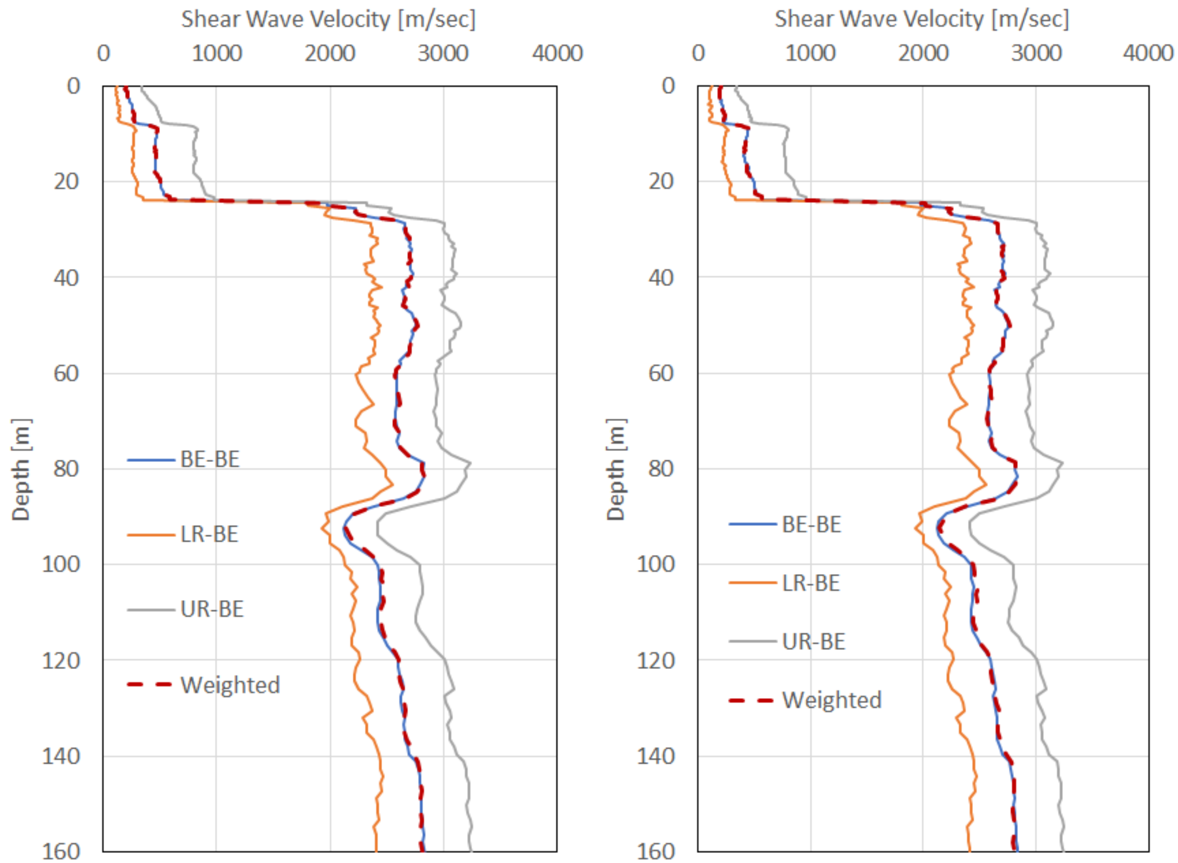


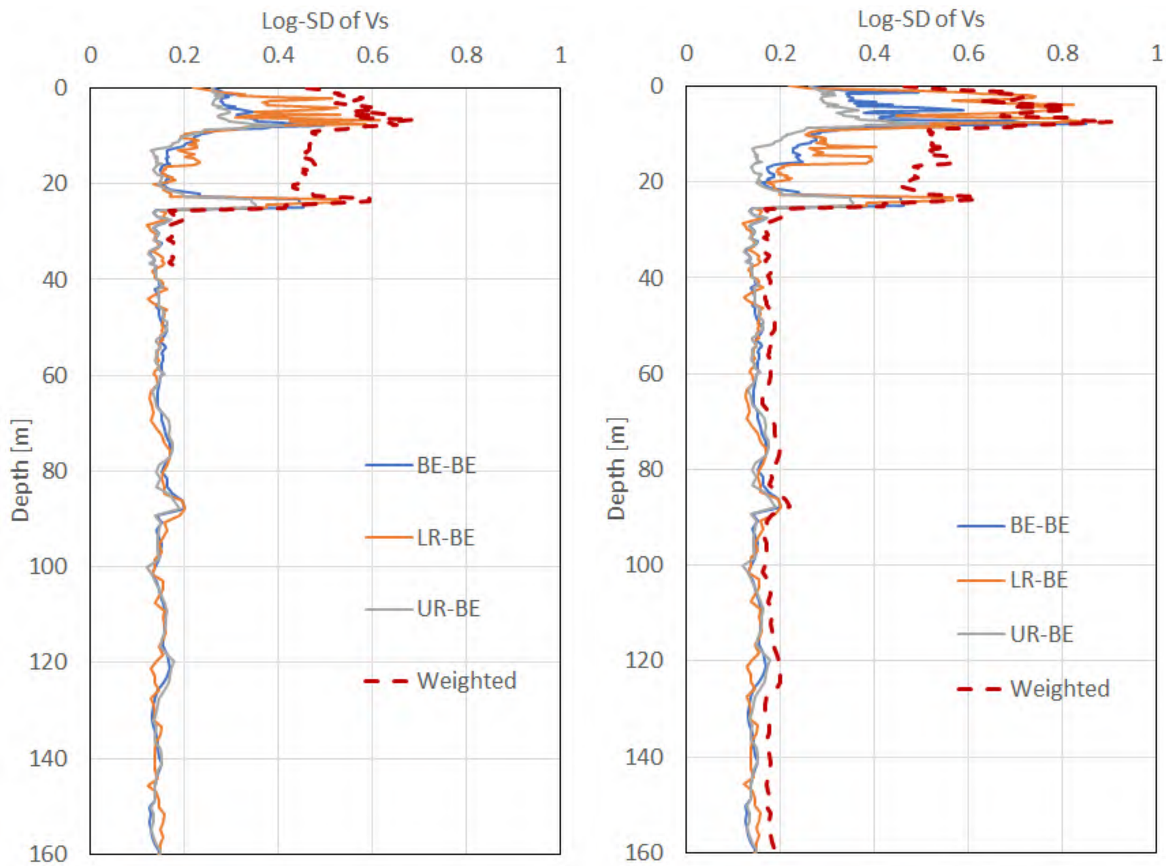
Figure 3.3-19: 5% Damped Response Spectra for Response Spectrum Matched HWA026 Acceleration Time Histories



A) 1 E-04 MAPE

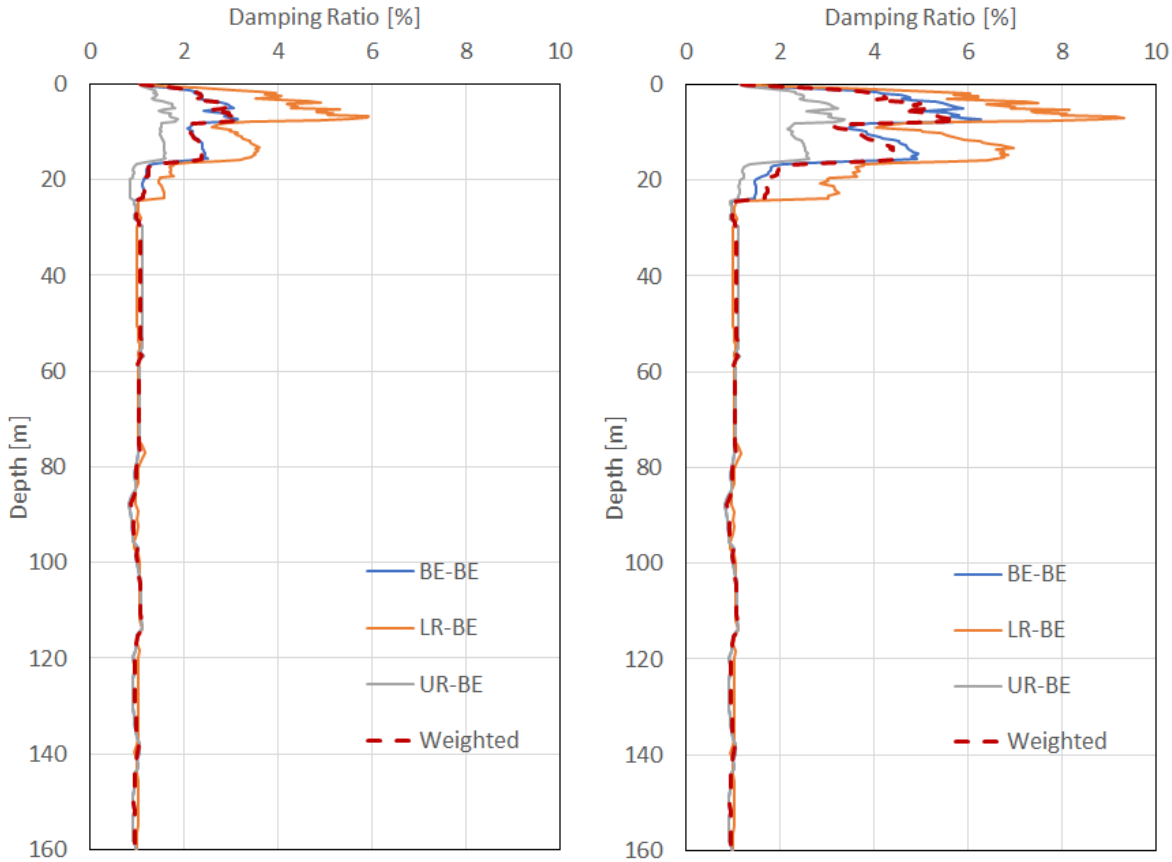
B) 1 E-05 MAPE

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



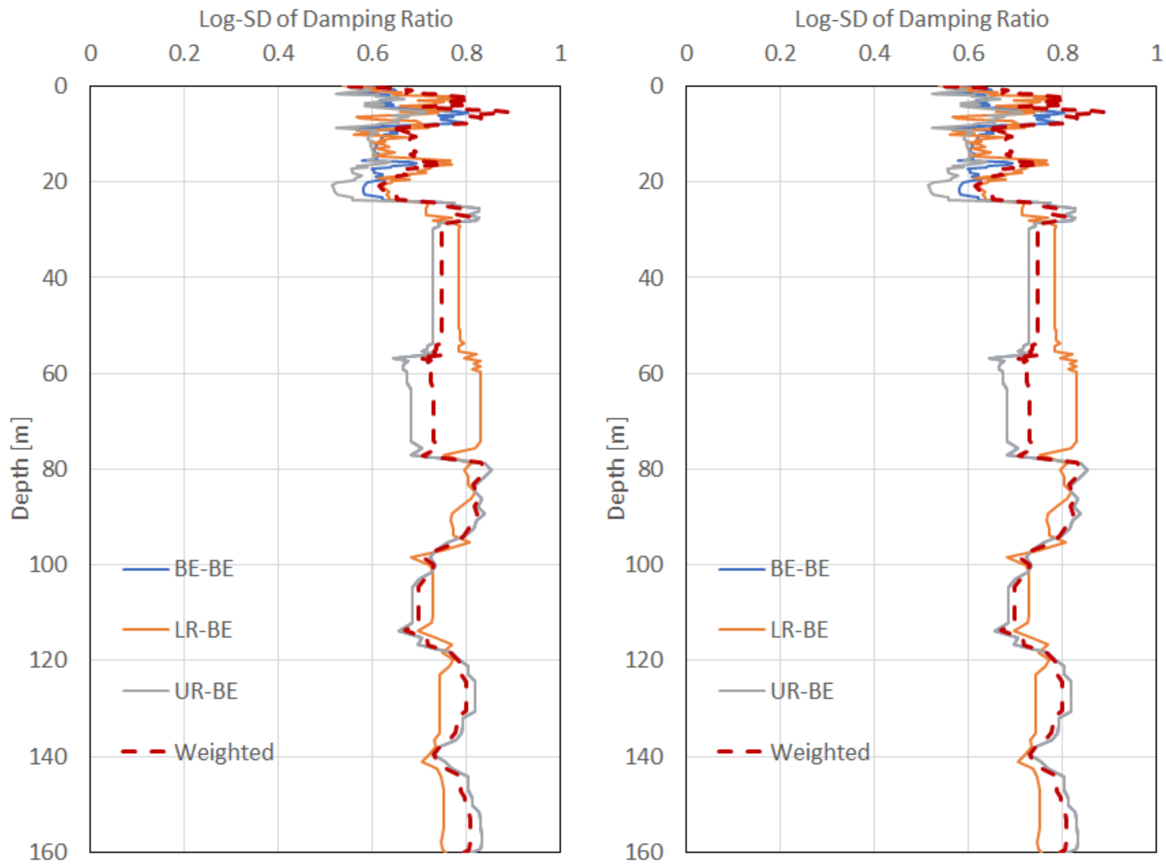
A) 1 E-04 MAPE B) 1 E-05 MAPE

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



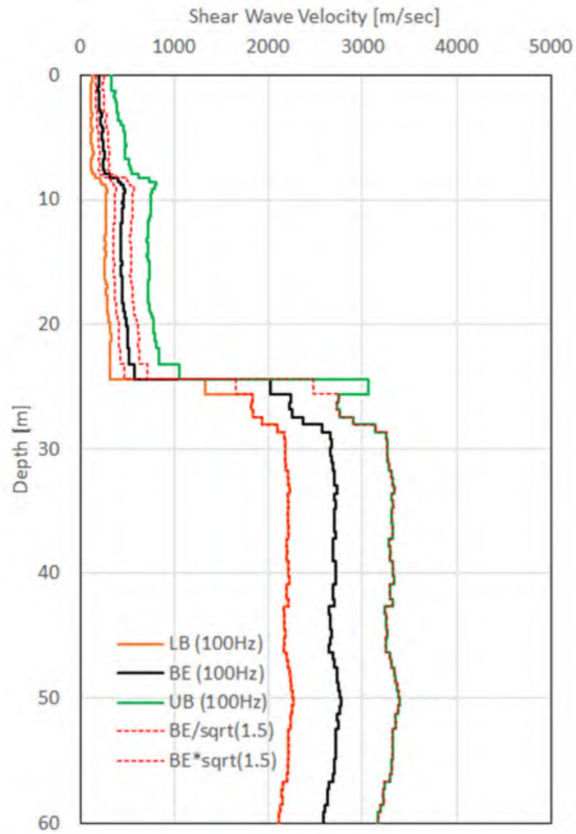
A) 1 E-04 MAPE B) 1 E-05 MAPE

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

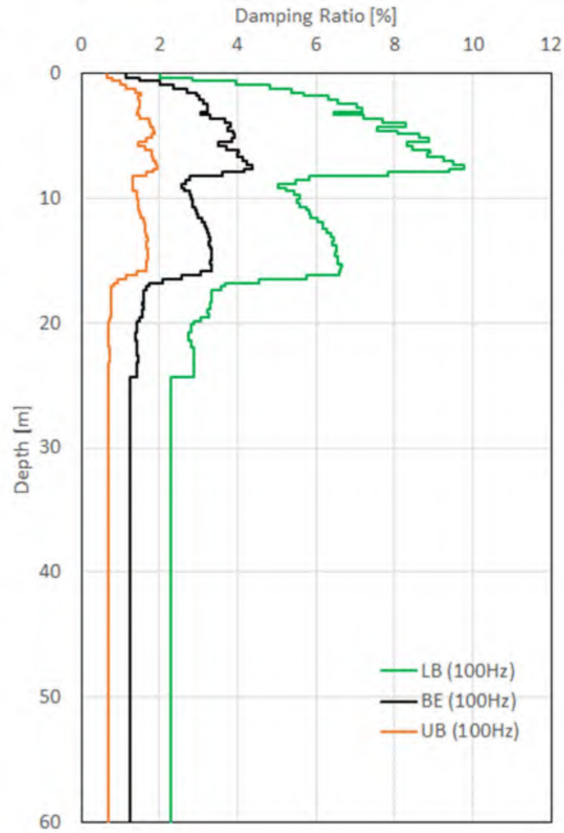


A) 1 E-04 MAPE B) 1 E-05 MAPE

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



a) Shear Wave Velocity



b) Shear Wave Damping Ratio

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

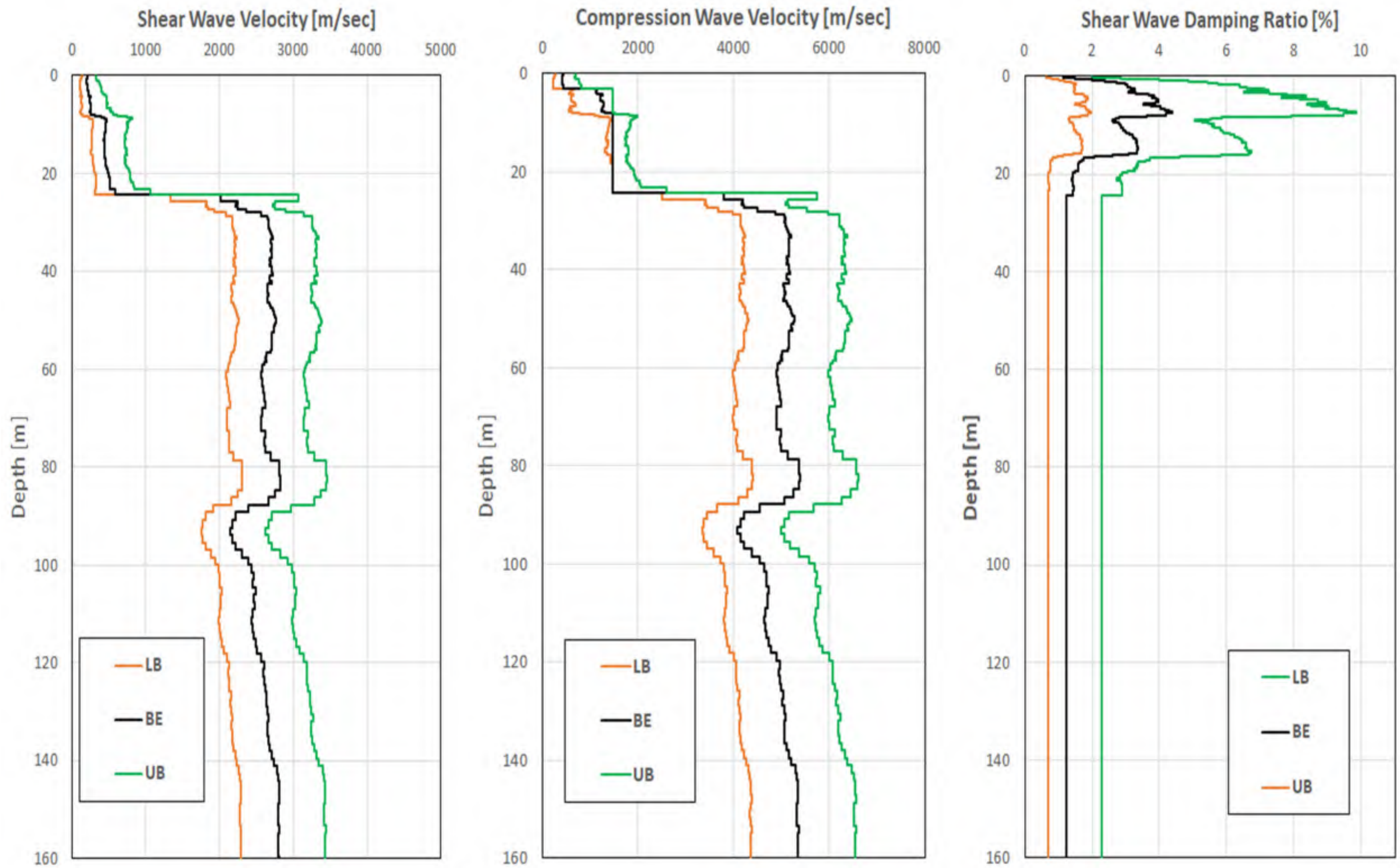


Figure 3.3-25: Subgrade Profiles for Bounding BWRX-300 Seismic Analyses

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:

- 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 -

listed or Factory Mutual approved boxing design. The required fire rating determines the fireproofing material thickness.

To satisfy requirements in Section 6.8.1.4 of CSA N293, wall and ceiling surface finishes are specified to meet flame spread index of 0-25 and smoke-developed index of 0-100 in accordance with CAN/ULS-S102 (Reference 3.4-8). Floor finishes have a flame spread rating of 0-300 and a smoke development classification less than 450 when tested in accordance with ASTM E648 (Reference 3.4-9) and ASTM E662 (Reference 3.4-10).

Suspended ceilings, including the lighting fixtures are of non-combustible construction in accordance with Section 5.7.1.1 of CSA N293.

To prevent the spread of spilled flammable and combustible liquids, including contaminated firefighting water, diking, draining or a combination of both is used to contain and control the volume of liquids in the buildings. Spill control measures are also included in the design to contain the contents of any above grade oil-filled vessel or tank larger than 208 liters and all tanks containing chemicals used in water/wastewater treatment or quality control.

3.4.1.1.2 General Protection Measures for Systems and Components

Complying with Section 6.8.4.1 of CSA N293, the BWRX-300 design minimizes the use of plastics, wood and other combustible materials in electrical equipment, cable raceways and wiring racks. Non-combustible and heat-resistant materials are used wherever practical throughout the unit.

Electrical cable in open tray raceways is limited to low voltage cable and meets IEEE 383 standards (Reference 3.4-11) in accordance with Section 6.8.4.4 of CSA N293. Vertical cables have a maximum vertical char of 1.5m when tested in accordance with the vertical flame tray test (Method 2-FT4) test in CSA C22.2 No. 2556 (Reference 3.4-12). Circuitry over 1000 volts is in conduit.

Certain areas of the plant have cable trays in stacked array. Where stacking of trays occurs, power cable, which is the most susceptible to internally generated fires, is routed in the uppermost tray to the greatest extent possible to provide isolation from other trays in the stack. A vertical separation is provided between horizontal cable trays. Groups of stacked trays for redundant SCN cables are separated horizontally.

Piping and cable tray penetrations are provided with fire-stops when penetrating fire rated barriers in accordance with Section 6.5.2.1 of CSA N293. Electrical cable fire-stops are tested to demonstrate a fire rating equal to the rating of the barrier they penetrate in accordance with Section 6.5.2.1 of CSA N293. As a minimum the penetrations meet the requirements of NUREG-1552 (Reference 3.4-13), including Supplement 1 of CSA C22.2 No 0.3 (Reference 3.4-14). The tests are performed or witnessed by a representative of a qualified, independent testing laboratory. The documented test results for the acceptable fire-stops are made a part of the plant design records.

To satisfy requirements in Section 6.3.1.1 of CSA N293, control, power, or instrument cables and equipment of redundant systems used for achieving and maintaining safe shutdown, are separated from each other by three hour rated fire barriers, except within inerted containment. Where the equipment of more than one division is required to be located within a single fire area (Control Room), cables are within conduit or a floor trench.

Fire separations are required to separate redundant fire safe shutdown systems and separate safe shutdown systems from other hazards.

Suitable design of the ventilation systems limits the consequences of a fire by preventing the spread of the products of combustion to other fire areas. Means are provided to ventilate, exhaust, or isolate the fire area as required, with consideration given to the consequences of ventilation system failure caused by the fire, resulting in a loss of control for ventilating, exhausting, or isolating a given fire area.

Filter media (excluding charcoal filters and High Efficiency Particulate Air (HEPA) filters) used in air handling systems meet the combustibility requirements of Class I in accordance with CAN/ULC-S111(Reference 3.4-15).

HVAC penetrations through 2-hour or 3-hour rated fire barriers are provided with fire/smoke dampers compatible with the rating of the fire barrier.

In accordance with Section 6.8.4.2 of CSA N293, electrical cabinets are designed to limit flame spread across cabinets.

3.4.1.2 Internal Explosions

The BWRX-300 fire hazard assessment evaluates the combustible loading along with the associated suppression requirements for each of the Power Block significant rooms and document the findings on the room data sheets.

Potential explosions of the following components are considered in the design:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen fires
- Offgas/hydrogen recombiners
- Transformers
- Transient combustibles
- Turbine auxiliaries

To satisfy requirements of CNSC REGDOC-2.5.2, Section 7.4.1, separation is provided between defense lines to the extent that defense lines are credited in the fault evaluation to mitigate the same event. Design measures considered include the use of fire barriers and blowout doors where flammable and combustible materials are located, and redundancy to enhance the reliability of systems.

Non-combustible and heat-resistant materials are also used, wherever practical throughout the Power Block, particularly in locations such as the containment and control rooms to reduce the risk of fires and explosions.

Administrative controls are also implemented to ensure stored chemicals and combustibles cannot ignite or react in sufficient quantities to impact nuclear safety. Collapse of structures, pipe whip, jet effects, and internal flooding as a result of internal explosions is also considered in the design.

3.4.1.3 Release of Internal Hazardous (Toxic) Gases

Plant personnel are protected from the adverse effects due to uncontrolled release of hazardous substances as a result of fires or internal explosions in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1, 7.12.1 and 7.12.2.

Preventive and mitigation measures against the release of hazardous and toxic gases include a proper design of ventilation systems to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Refer to Chapter 9A, Sections 9A.5 and 9A.6 for details of the BWRX-300 HVAC and fire protection systems, respectively.

Complying with CNSC REGDOC-2.5.2, Sections 8.10.1 and 8.10.2, the habitability of the MCR and SCR is ensured by designing the HVAC systems in these rooms to detect and limit the introduction of airborne radioactivity, toxic gas or smoke into the rooms as described in Chapter 6, Section 6.4. As stated in Chapter 6, Section 6.4.2.1, habitability requirements in the control rooms are maintained without credit for any breathing apparatus or protective clothing.

HVAC systems also supply outside air into the SCCV via the containment inerting system and exhaust inerting gases to provide a habitable environment for maintenance personnel during outage and maintenance periods.

3.4.2 Internal Flooding

SC1 SSC and SC2/SC3 SSC credited with flood event mitigation in the fault evaluation are protected against internal flooding in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1.

Appropriate means are included in the design to prevent failure of SSC that are not designed to be submerged or exposed to spray as a result of flooding. They include the use of redundant system trains or divisions, structural barriers or compartments, curbs and elevated thresholds, and a leak detection system.

The design of the integrated RB structures considers the loads associated with the post-accident internal flooding of the containment following a DBA. The hydrostatic loads from the maximum possible water level are applied as pressures to the affected walls and mat foundation and applicable loads are also used for design of containment metal components.

The BWRX-300 internal flooding analysis identifies flooding sources, equipment in each area, and maximum internal flood levels in each area. The sources of internal flooding hazards include:

- Leaks and breaks in pressure retaining components
- High-energy piping breaks and cracks
- Moderate-energy piping through-wall cracks
- Pump mechanical seal failures
- Failure of isolating devices
- Storage tank ruptures
- Actuation of fire protection system
- Flow from upper elevations and nearby areas

The flood level in each internal area is determined by evaluating the inflow due to internal flooding sources, outflow from area compartment, and accumulation in each compartment area due to net flow.

3.4.3 Internal Missiles

Complying with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1, the BWRX-300 design includes preventive and mitigation measures against internal missiles. The methodology used to determine internal missiles is discussed in Subsection 3.4.3.1, while Subsection 3.4.3.2 provides the general preventive and mitigation measures considered in the design.

3.4.3.1 Sources of Internal Hazards

Potential missiles inside and outside containment and turbine missiles are identified, and their statistical significance determined. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or exceedance of radiological release limits. Criteria for determining statistically significant missiles are obtained from applicable portions of U.S. NUREG-0800 (Reference 3.4-16), SRP 3.5.1.1 through 3.5.1.3.

These missile sources could result from in-plant component overspeed failures or high-pressure system ruptures in compliance with CNSC REGDOC-2.5.2, Section 7.4.1. Rotating equipment failures include evaluations of pumps, fans, blowers, diesel generators, compressors, and turbines. Potential missiles from failure of pressurized components include valve bonnets, valve stems, pressure vessels, thermowells, retaining bolts, and blowout panels.

3.4.3.2 Protection from Internal Missile Hazards

Preventive and mitigative measures considered in the BWRX-300 design against internal missiles include the following:

- Locating the system or component in an individual missile-proof structure
- Physically separating redundant systems or components of the system from the missile trajectory path or calculated range
- Providing localized protection shields or barriers for systems or components
- Designing the particular structure or component to withstand the impact of the most damaging missile
- Providing design features on the potential missile source to prevent missile generation
- Orienting the potential missile source to prevent unacceptable consequences caused by missile generation

Refer to Subsection 3.3.5.4 for barrier design procedures for impactive loads, including internal missiles.

3.4.4 Pipe Breaks

BWRX-300 SC1 SSC and SC2/SC3 SSC credited with event mitigations in the fault evaluation are adequately protected from the consequences associated with a postulated rupture of high-energy piping and crack of moderate-energy piping inside and outside containment in compliance with Sections 7.4.1 and 7.7 of CNSC REGDOC-2.5.2 and IAEA SSG-64. Design bases and measures used to protect these SSC, referred to in the following subsections as essential SSC, are discussed in Subsections 3.4.4.1 and 3.4.4.2.

Effects that may result from a postulated rupture of high-energy piping include (1) pipe whipping, (2) pipe break reaction forces, (3) jet impingement forces, (4) blast waves, (5) sub-compartment pressurization, (6) decompression waves, (7) Missile generation, (8) environmental effects and (9) Flooding.

In the BWRX-300 design, a whipping pipe may hit a target and cause secondary failure in the target object depending on the thrust force, materials and sizes of the pipe/target. Severance in the target may occur and form a missile. A pipe whipping about a plastic hinge is not assumed to cause severance at the plastic hinge. Therefore, a break cannot cause the whipping pipe to act as a missile. Criteria related to the evaluation of and protection against missiles, including those resulting from jet impingement or a whipping pipe, are provided in Subsection 3.4.3.

Protection against flooding and environmental effects as a result of high-energy pipe breaks are discussed in Subsections 3.4.2 and 3.9.4, respectively.

3.4.4.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.4.4.1.1 Design Basis

In addition to meeting requirements in CNSC REGDOC-2.5.2 and IAEA SSG-64, the BWRX-300 pipe break event protection also conforms to 10 CFR 50 Appendix A (Reference 3.4-17), General Design Criterion 4. To supplement the guidance provided in IAEA SSG-64, the design bases for this protection are in compliance with NRC Branch Technical Position (BTP) 3-3 (Reference 3.4-18) and BTP 3-4 (Reference 3.4-19) included in Subsections 3.6.1 and 3.6.2, respectively, of U.S. NUREG 0800. BTP 3-4 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

1. Assure that the reactor can be shut down safely and maintained in a safe shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits with Loss of Preferred Power (LOPP).
2. Assure that containment integrity and leak tightness are maintained.

3.4.4.1.2 Design Evaluation

An analysis of pipe break events is performed to identify those essential systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the pipe break event.

Pipe break events involving high-energy fluid systems are evaluated for the effects of pipe whip, jet impingement, flooding, sub-compartment pressurization, and other environmental effects. Pipe break events involving moderate-energy fluid systems are evaluated for wetting from spray, flooding, and other environmental effects.

Adequate protection is provided against the effects of pipe break events for essential SSC to an extent that their ability to shut down the plant safely or mitigate the consequences of the postulated pipe failure is not impaired. This is accomplished by means of design features such as physical separation, jet shields and pipe whip restraints or by designing the SSC to accommodate applicable loads due to postulated pipe failure.

3.4.4.1.3 General Protection Measures

The direct effects associated with a particular postulated break or crack are mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the following specific measures for protection against actual pipe movement and other associated consequences of postulated failures:

1. Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

2. As an alternative to protective measures, SSC identified as essential targets under postulated pipe breaks are analyzed to show that the essential functionality remains available under all applicable loading conditions resulting from the pipe break.
3. The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.
4. Protection of SCN systems and components from the effects of postulated pipe breaks is considered where a resulting failure of the SCN system or component could lead to failure of an essential SSC. This includes consideration of coatings and insulation materials which could result in debris generation

Separation

To meet requirements in CNSC REGDOC-2.5.2, Section 7.6.1.1, the plant layout arrangement provides physical separation and segregation of essential SSC to the extent practicable to provide sufficient distance such that the effects of the failure cannot impair their essential functionality.

Physical separation between redundant safety class systems supporting Defense Line 3 (DL3) with their related auxiliary supporting features is another basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

Pipe Whip Restraints

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, analysis of the SSC or enclosures alone, and when it is necessary to limit the piping movement (pipe whip) following a postulated break. Restraints are located based on the specific postulated break locations determined in accordance with Subsection 3.4.4.2. After the restraints are placed, the piping and essential SSC are evaluated for jet impingement and pipe whip. For those cases where unacceptable jet impingement damage could still occur, barriers, shields, or enclosures are utilized in conjunction with pipe whip restraints.

The design criteria for restraints are given in Subsection 3.4.4.2.

Barriers, Shields, and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present because of spatial separation or existing plant features, additional barriers, deflectors, shields, or guard pipes are provided as necessary to meet the functional protection requirements of essential targets.

Structures acting as barriers, shields, or enclosures are designed to withstand the consequences of postulated pipe failures (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with other internal hazards such as missiles and loadings associated with the DBE within their respective design load limits. Procedures used to design these structures are provided in Subsection 3.3.5.4.

The BWRX-300 barrier design ensures a resistance to impulsive loads that is at least 20% greater than the steady-state magnitude of the impulsive load in accordance with regulatory guidance of U.S. NRC RG 1.243 (Reference 3.4-20), Regulatory Position 11.1.2 and provisions of CSA N291, Clause A.3.5.1.

3.4.4.1.4 Protective Features and Operator Actions

All available systems are considered for mitigating the consequences of a failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such as unit trip and LOPP, and of the assumed single active component failure and its direct consequences.

As stated in Chapter 15, Section 15.5, no operator actions are required to mitigate the effects of high-energy pipe breaks.

3.4.4.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section discusses the location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate - energy fluid system piping inside and outside of the primary containment. This information provides the design basis for the requirements for protection of essential SSC.

3.4.4.2.1 Criteria Used to Define Break and Crack Location and configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

Definition of High-Energy Fluid Systems

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 93.3°C; and
- Maximum operating pressure exceeds 1.9 MPaG.

Definition of Moderate-Energy Fluid Systems

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where either or both of the following are met:

- Maximum operating temperature is 93.3°C or less; and
- Maximum operating pressure is 1.9 MPaG or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are limited to postulation of cracks in piping and branch runs; these cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for conservative environmental conditions in a confined area where high and moderate-energy fluid systems are located.

The following high-energy piping systems are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from damage effects:

- Main Steam
- Isolation Condenser System
- Control Rod Drive System
- Reactor Water Cleanup System
- Condensate Feedwater System
- Condenser Offgas System (in TB)

Moderate-Energy piping systems considered as potential candidates for a postulated pipe crack include the following:

- Boron Injection
- IC Pool Cooling
- Shutdown Cooling
- Fuel Pool Cooling
- Passive Containment Cooling
- Containment Inerting

3.4.4.2.2 Location of Postulated Pipe Breaks

Postulated pipe breaks are selected as follows:

Piping in Containment Penetration Areas

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated breaks. .

ASME Code Section III Class 1 High-Energy Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 1 piping (Reference 3.4-21) are postulated at the following locations in each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress range or fatigue usage values exceed the limits specified in BTP 3-4

ASME Code Section III Class 2 and 3 High-Energy Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 2 and 3 piping (Reference 3.4-22) are postulated at the following locations in those portions of each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress values exceed the limits specified in BTP 3-4

Non-ASME High-Energy Piping

Breaks in seismically analyzed non-ASME high-energy piping systems are postulated according to the same criteria as for ASME Code Section III, Class 2 and 3 high-energy piping systems.

Breaks in non-seismically analyzed, non-ASME high-energy piping systems are postulated at each terminal end and at each intermediate location of potential high stress or fatigue, such as pipe fittings, valves, flanges, and welded-on attachments

3.4.4.2.3 Location of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

Piping in Containment Penetration Areas

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated cracks.

High-Energy Piping in Areas Other Than Containment Penetrations

With the exception of those portions of piping identified above as containment penetration areas, cracks in high-energy piping are postulated as follows:

1. For ASME BPVC Code, Section III Class 1 piping, at axial locations where the calculated stress range values exceed the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress values exceed the limits specified in BTP 3-4.
3. For piping which has not been evaluated to obtain stress information, through-wall cracks are postulated at axial locations that produce the most severe environmental effects.

Moderate-Energy Piping in Areas Other Than Containment Penetrations

With the exception of those portions of piping identified above as containment penetration areas, through-wall cracks in moderate-energy piping adjacent to safety class SSC are postulated except where:

1. For ASME BPVC Code, Section III, Class 1 piping the calculated stress range values are less than the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III, Class 2 or 3 and non-ASME class piping, the calculated stress values are less than the limits specified in BTP 3-4.

Through-wall cracks, unless the piping system is exempted above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.

Through-wall cracks are postulated in fluid system piping designed to non-seismic standards as necessary to assure that essential system and component functionality is maintained following a piping failure assuming a concurrent single active failure.

Moderate-Energy Piping in Proximity to High-Energy Piping

In cases where both high-energy and moderate-energy piping systems exist in a confined area, cracks are postulated in the piping system which leads to the more conservative environmental conditions.

3.4.4.2.4 Types of Breaks and Cracks to be Postulated

Pipe Breaks

The following criteria are used to postulate breaks in high-energy fluid system piping at the identified locations:

1. For the purposes of considering dynamic effects, circumferential breaks are postulated only in piping having a nominal diameter greater than 25 mm.
2. Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than 100 mm.
3. Longitudinal breaks are not postulated at terminal ends.
4. Circumferential breaks are assumed at all terminal ends.
5. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsection 3.4.4.2.2, consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location is used to identify the most probable type of break based on the BTP 3-4 rules.
6. Where breaks are postulated to occur at each intermediate pipe fitting, weld attachment, or valve without the benefit of stress calculations, only circumferential breaks are postulated.
7. For a circumferential break, the dynamic force of the jet discharged at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient.
8. For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location.

Pipe Cracks

The following criteria are used to postulate through-wall leakage cracks in high- or moderate-energy fluid system piping at the identified locations:

1. Leakage cracks are only postulated in piping having a nominal diameter greater than 25 mm.
2. The postulated cracks are oriented circumferentially to result in the most severe environmental consequences.
3. Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half-pipe-diameter in length and one-half-pipe-wall thickness in width.
4. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.

3.4.4.2.5 Analysis Methods to Define Blowdown Forcing Functions and Response Models

Analytic Methods to Define Blowdown Forcing Functions

Analytical methods used to establish pipe rupture blowdown and jet thrust forcing forces are in accordance with ANSI/ANS 58.2 (Reference 3.4-23), Section 6.2.

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors.

Criteria used for calculation of fluid blowdown forcing functions include the following:

1. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).
2. For a circumferential break, the dynamic force of the jet discharge at the break location is based on the cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.
3. All breaks are assumed to attain full size within one millisecond after break initiation.

Pipe Whip Dynamic Response Analysis Criteria

Dynamic forces are assumed to cause pipe whip reaction whenever moments cause excessive plastic deformation and the formation of a plastic hinge. Significant motion occurs only when the thrust force acts through an arm of sufficient length to induce a plastic hinge. This length is called the plastic hinge length. When the stiffness of a piping system is such that a plastic hinge cannot form, the pipe lateral displacement is assumed to be equal to the pipe diameter.

Pipe whip restraints are used to prevent piping from deforming plastically by forming hinges. They absorb blowdown force energy and limit jet impingement’s zone of influence.

The prediction of time dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from ruptured pipe is used as an input to evaluate the pipe whip dynamic response.

Pipe motion following circumferential breaks are assumed in the plane defined by the initial axis of the jet thrust force and rotation about a plastic hinge point, or at an intermediate point, such as the second change in direction, where the moment resisting capacity is less than straight pipe, provided the distance to this point is not significantly less than the plastic hinge length. The arc of the whipping pipe for planar motion is assumed to be limited to 180 degrees due to crimping at the plastic hinge and the pipe folding back against itself. Where a system consisting of piping, restraints and supporting structures is so complex that the assumption of planar motion is neither conservative nor realistic, the whip zone of influence can be conservatively enlarged to a region approaching a sphere with a radius equal to the distance between the break point and the first restraint. In lieu of this assumption, a more detailed elastoplastic analysis may be performed.

Longitudinal breaks in the form of axial split without pipe severance are postulated in the centre of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping configuration and produces out-of-plane bending.

Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

For restrained longitudinal breaks or those breaks for which it can be shown that the pipe resists bending elastically, the zone of whip influence is taken to be all points within a distance of one pipe diameter from the axis of the pipe, unless physically limited by piping restraints, structural members or piping stiffness. For unrestrained longitudinal breaks in elbow fittings, the out-of-plane forces are assumed to cause whipping through a zone of influence described by the rotation of the fitting through 360 degrees about an axis which connects the two plastic hinges formed in the attached legs of piping.

A whipping pipe is considered capable of rupturing impacted pipes of smaller nominal pipe diameter, and of developing through-wall cracks in impacted pipes of equal or larger nominal pipe sizes with thinner wall thickness.

If a whipping pipe contains a large in-line mass (such as a valve), or if there is a change in the pipe shape (e.g., an elbow) near the end of the pipe, rupture of target pipes which are equal to or larger than the whipping pipe is considered.

Pipe Whip Dynamic Response Methods

Analytical models used to evaluate pipe whip dynamic response adequately represent the mass, inertia and stiffness properties of the piping system accounting for interaction effects of both the piping and pipe whip restraint.

Analytical methods used for piping response are based on those defined in ANSI 58.2, Section 6.3 and include complete system dynamic analysis, simplified dynamic analysis, quasi-dynamic analysis, energy balance analysis, and static analysis.

In cases where it is necessary to calculate stresses at locations which are far away from the break (e.g., in containment penetration break exclusion area), a more extensive model of the ruptured piping, supports, and pipe whip restraints is necessary.

If the snubbers or other seismic restraints are included in the piping model, they are modeled with the same stiffness used in the seismic analysis of the pipe. However, credit for seismic restraints cannot be taken if the applied load exceeds the ASME BPVC Code Section III (Reference 3.4-21, Reference 3.4-22 and Reference 3.4-24) Service Level D rating.

Pipe Whip Analysis Material Properties

Strain rate effects and other material property variations are considered in the pipe whip analysis of piping and pipe whip restraints.

Material properties and design limits consistent with those stated in ANSI/ANS 58.2, Sections 6.6.2 and 6.6.3 are applied for plastic deformation design of piping and pipe whip restraint design under dynamic and steady-state loading conditions.

3.4.4.2.6 Dynamic Analysis Methods to Verify Integrity and Operability

Jet Impingement Analyses and Effects on Essential Components

For each postulated circumferential and longitudinal break, an evaluation of jet impingement effects on essential targets including jet impinging force, thermal energy, and moisture is completed in accordance with the methodology criteria in this section.

In the case of circumferential breaks, jets are assumed to be oriented axially with respect to the pipe. In the case of longitudinal breaks, jets are assumed to be oriented radially.

Potential targets, or portions of targets adjacent to the jet boundary, are assumed to be impinged upon when reasonable variations in jet geometry or pipe movement are considered.

In evaluating the potential for jet impingement on specific targets, consideration is given to the movement of the jet centreline due to pipe whip, including pipe-restraint interaction.

Thermal and moisture effects on essential targets are determined in accordance ANSI/ANS 58.2, Section 7.4 and 7.5.

Modeling of the jet geometry and determination of the jet impingement force acting on a target is calculated according to ANSI/ANS 58.2, Sections 7.2, 7.3, and Appendices C and D, with modifications applied as identified in NUREG/CR-7275 (Reference 3.4-25).

Pipe Whip Effects on Essential Structures, Systems and Components

This section provides the criteria and methods used to evaluate the effects of pipe displacements on essential SSC following a postulated pipe rupture.

Pipe whip (displacement) effects on essential SSC can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurs in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays and conduits.

(1) Pipe Displacement Effects on Components in the Same Piping Run

Essential components located in the same run as the postulated break meet the applicable ASME Code class limits for Service Level D and limits to ensure required operability.

(2) Pipe Displacement Effects on Essential Structures, Systems, and Components

The criteria and methods used to calculate the effects of pipe whip on external components consist of the following:

1. The effects on barriers, shields, or enclosures credited for protecting essential SSC are evaluated in accordance with the barrier design procedures given in Subsection 3.3.5.4.
2. If the whipping pipe impacts an essential system or component, mitigating measures are established to ensure essential functionality is not lost for the postulated break scenario.

Loading Combinations and Design Criteria for Pipe Whip Restraint

Pipe whip restraints are non-ASME code class components. As a result, other methods (i.e., testing) such as the use a reliable database may be used instead of the rules applied to ASME code class components for their design and sizing.

Pipe whip restraints are designed for both the thrust force at the pipe rupture location and the impact force of the pipe. The magnitude of these forces is a function of the pipe size, fluid temperature, and operating pressure.

Pipe whip restraints, as differentiated from piping supports, are typically designed only to function, and carry loads for an extremely low probability gross failure in a piping system carrying high-energy fluid. They are also required to remain functional following an earthquake up to and including the design basis DBE.

Pipe whip restraints are designed with sufficient clearances to prevent an increase in the pipe stresses by their presence during any normal mode of reactor operation or condition and are designed to allow for in-service inspection of the process piping with minimal obstruction.

3.4.4.2.7 Analytic Methods to Define Blast Wave Interaction to SSC

Sub-compartment pressurization due to postulated pipe breaks is considered where applicable.

3.4.4.2.8 Sub-compartment Pressurization

As discussed in Chapter 6, Subsection 6.3.2.2, the BWRX-300 containment sub-compartments do not contain large high-energy pipes and are, therefore, not subject to sub-compartment pressurization loads. For breaks outside the containment, mass and energy releases into the sub-compartments are calculated as described in Chapter 15, Subsection 15.5.9.2. Pressurization of the sub-compartments of the reactor building is calculated using the GOTHIC code described in Chapter 15, Subsection 15.5.1.2. The GOTHIC model of the RB includes all sub-compartments of the RB as lumped parameter volumes, including all flow passages between the rooms. This includes all doors and blowout panels which may be closed normally but may open if a pressure differential develops between the sub-compartments.

3.4.4.2.9 Decompression Waves

3-D thermal hydraulic code TRACG (See Chapter 15, Subsection 15.5.1.2) generates pressure time history in the annular region between chimney/shroud and RPV due to acoustic decompression wave as a result of a pipe break. Generated time history is part of the inputs to RPV primary structural FE model along with jet impingement, jet reaction and pipe whip restraint loads inputs to determine dynamic effects on RPV components, RPV internals and nozzles/pipings attached to RPV.

3.4.5 Other Internal Hazards

3.4.5.1 Hard Object Impact

Complying with CNSC REGDOC-2.5.2, Section 7.15.3 and IAEA SSG-64, the BWRX-300 design considers hard object impact loads resulting from the drop of heavy loads lifted and handled in areas where SSC required for safe shutdown of the plant are located.

Drops considered are those most likely to occur during the handling of plant equipment for maintenance or during spent fuel transfer operations. Other drops considered are drops as secondary effects of other internal hazards or external hazards discussed in Section 3.3.

In accordance with U.S. NRC RG 1.244 (Reference 3.4-26), the BWRX-300 heavy load is defined per the provisions of U.S. NUREG-0612 (Reference 3.4-27) as any load, carried in a given area after a plant becomes operational, that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Critical heavy load handling evolutions considered are those where inadvertent operations or equipment malfunctions, separately or in combination, could:

- Cause a release of radioactivity
- Cause a criticality accident
- Cause the inability to cool fuel within the reactor vessel or within the Fuel Pool
- Prevent a safe shutdown of the reactor

Measures considered to reduce the potential of heavy load drops in the RB meet the D-in-D guidelines in U.S. NRC RG 1.244 and Section 5.1 of US NUREG-0612. They include a proper plant arrangement, the implementation of a heavy loads program as part of the plant procedures and effective means of lifting and transporting heavy loads designed to satisfy the single failure proof guidelines of Section 5.1.6 of US NUREG-0612.

Chapter 9A, Subsection 9A.8.1 provides an overview of the BWRX-300 heavy load program which identifies all heavy loads lifted during operation of the plant and the safe travel paths determined for their lifting. This program also manages the safe execution of heavy load evolutions.

Chapter 9A, Subsection 9A.8.1 describes the various cranes and hoists used to lift and transport heavy loads and applicable guides and standards used for their design. The RB polar crane main and auxiliary hoists meet the requirements of single failure proof systems in accordance with ASME NOG-1 (Reference 3.4-28). The refueling platform main hoist meets the requirements of a single failure proof hoist. Periodic inspection and maintenance of cranes are also planned to ensure their safe functioning.

3.4.5.2 Failure of Non-Structural Element

The failure of non-structural elements is considered in the BWRX-300 design.

Staircases and elevator shafts are evaluated and designed for interaction with plant Seismic Category A or B SSC in the event of DBE.

Architectural components and shielding blocks whose failure or dislocation could affect the safe operation of any Seismic Category A or B SSC are also evaluated for seismic interaction.

Scaffolding and other temporary structures considered a temporary alteration in support of maintenance are evaluated for seismic interaction as well, following the plant temporary structures procedure.

3.4.5.3 Electromagnetic Interference

Internal electromagnetic interference is caused by induction or radiation from installed equipment.

Complying with CNSC REGDOC-2.5.2, Section 7.5, safety class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

Qualification requirements for protection against electromagnetic interference are presented in Subsection 3.9.5.

Plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements are discussed in Chapter 8, Section 8.6.

3.4.6 References

3.4-1 CNSC Regulatory Document REGDOC-2.5.2, “Design of Reactor Facilities: Nuclear Power Plants.”

3.4-2 IAEA Safety Standards Series No. SSG-64, “Protection against Internal Hazards in the Design of Nuclear Power Plants,” International Atomic Energy Agency.

3.4-3 IAEA NS-G-1.11, “Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants,” International Atomic Energy Agency.

3.4-4 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.

3.4-5 CSA N293, "Fire Protection for Nuclear Power Plants," CSA Group.

3.4-6 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors),” CSA Group.

3.4-7 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.

3.4-8 CAN/ULC-S102, “Method of Test for Surface Burning Characteristics of Building Materials and Assemblies,” Underwriters’ Laboratories of Canada.

3.4-9 ASTM E648, “Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source,” American Society for Testing and Materials.

3.4-10 ASTM E662, “Standard Test Method for Specific Optical Density of Smoke Generated by Solid Materials,” American Society for Testing and Materials.

3.4-11 IEEE 383-2015, “IEEE Standard for Qualifying Electrical Cables and Splices for Nuclear Facilities,” Institute of Electrical and Electronic Engineers.

3.4-12 CAN/CSA C22.2 No. 2556, “Wire and Cable Test Methods,” CSA Group.

3.4-13 USNRC NUREG-1552, “Fire Barrier Penetration Seals in Nuclear Power Plants.”

3.4-14 CAN/CSA C22.2 No 0.3-09, “Test Methods for Electrical Wires and Cables,” CSA Group.

3.4-15 CAN/ULC-S111-13, “Standard Methods of Fire Tests for Air Filter Units,” Underwriters’ Laboratories of Canada.

3.4-16 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,"

3.4-17 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants.”

3.4-18 USNRC BTP 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.”

3.4-19 USNRC BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.”

3.4-20 USNRC Regulatory Guide 1.243, “Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments.”

3.4-21 ASME BPVC-III NB, "Section III - Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.

3.4-22 ASME BPVC-III NCD, “BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components,” American Society of Mechanical Engineers.

3.4-23 ANSI/ANS 58.2-1988, “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture,” American National Standards Institute/American Nuclear Society.

3.4-24 ASME BPVC-III NE-2021, “BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components,” American Society of Mechanical Engineers.

3.4-25 USNRC NUREG/CR-7275, “Jet Impingement in High-Energy Piping Systems.”

3.4-26 USNRC Regulatory Guide 1.244, “Control of Heavy Loads at Nuclear Facilities.”

3.4-27 USNRC NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.”

3.4-28 ASME NOG-1, “Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” American Society of Mechanical Engineers.

3.5 General Design Aspect for Civil Engineering Works of Seismic Category Buildings and Civil Engineering Structures

This Section presents the design principles, design basis requirements, criteria and applicable codes and standards used in the design of the BWRX-300 civil structures, including their foundations in compliance with requirements in CNSC REGDOC-1.1.2 (Reference 3.5-1), Section 4.5.5.

Below are the key PSAR sections that impact the BWRX-300 Civil/structural design that should be reviewed along with this section:

- Chapter 1 which provides the DNNP general site and facility layout, a description of the BWRX-300 buildings, plant operational modes, principles of safety management and applicable codes & standards utilized in the design
- Chapter 2 which described the characteristics of the DNNP site on which the BWRX-300 facility is constructed
- Chapter 3, Section 3.1, which provides the general design aspects and D-in-D safety framework utilized in the BWRX-300 design
- Chapter 3, Section 3.2, which provides the general classification of BWRX-300 SSC and the approach used to establish these classifications
- Chapter 3, Sections 3.3 and 3.4, which provide methodology and general design requirements for protection against the effects of external and internal hazards
- Chapter 9B which provides specific information on compliance with the design rules for civil engineering works and structures

From the site layout presented in Chapter 1, Appendix A, Figure A1.4-1, the primary buildings in the BWRX-300 Power Block consist of the Reactor Building (RB) which houses the containment, Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), and Reactor Auxiliary Bay. In the following sections, reference to the integrated RB structure is inclusive of the RB, containment, and containment internal structures, whereas RB is used to refer to the part of the integrated structure located outside of containment.

The seismic categorization of these structures is provided in Table 3.3-1. Per Subsection 3.2.3 and Table 3.3-1, the Seismic Category A integrated RB housing SC1 SSC has the utmost importance to safety and is credited for the safety analysis of the BWRX-300. RWB structures that support and protect equipment and components for storage and processing of highly radioactive gas, liquids and solid materials are categorized as RW-IIa. The CB, TB and Reactor Auxiliary Bay categorized as Non-Seismic structures are not credited in the safety analysis but are relied upon for their D-in-D function since they house and protect SC2 or SC3 systems and components. The RWB, CB, TB, and Reactor Auxiliary Bay can also affect the BWRX-300 safety considering their proximity to and interaction with the integrated RB structure.

Other civil structures for which design basis requirements are provided are the Pumphouse/Forebay structures and tunnels that support the condenser cooling and plant cooling water systems, and the Fire Pump Enclosure. For the location of these structures, refer to Chapter 1, Appendix A, Figure A1.4-1.

In accordance with Section 3.1 of CNSC REGDOC-1.1.5 (Reference 3.5-2) and Section 5.4 of CNSC REGDOC-3.5.3 (Reference 3.5-3), design principles for BWRX-300 structures are provided in a graded manner commensurate to their importance to safety. The primary focus of this Section is for the Seismic Category A integrated RB. Design principles for the RWB, CB, TB, Reactor Auxiliary Bay, Pumphouse/Forebay and Fire Pump Enclosure structures are provided in Chapter 9B, Section 9B.3.

Remaining plant structures shown in Chapter 1, Appendix A, Figure A1.4-1 are not covered since they are not credited in the safety analysis.

3.5.1 General Design Principles for Seismic Category A Structures

The BWRX-300 Seismic Category A integrated RB structure is designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, Anticipated Operational Occurrence (AOO) and DBA in compliance with requirements in CNSC REGDOC-2.5.2 (Reference 3.5-4), Sections 7.15.1 and 7.7. The robustness of the design to prevent potential release of radioactivity to the public and environment under Design Extension Condition (DEC) is considered in compliance with requirements in CNSC REGDOC-2.5.2, Sections 7.7 and 7.15.1 and is discussed in Subsection 3.5.6.

The integrated RB structure and its common foundation are primarily constructed using an advanced steel-plate composite system called Steel Bricks™. The Steel Bricks™ system has a configuration similar to the typical steel-plate composite system except that the tie-rods in the typical steel-plate composite system are replaced by diaphragm plates created by bending the plates that facilitates the fabrication process. The Steel Bricks™ modules used to construct the integrated RB comprise of a pair of steel faceplates, shear connectors, diaphragm plates, and concrete fill. The faceplates and concrete fill act as the composite system to provide strength and stability to the Steel Bricks™ system. The shear connectors facilitate the composite action between the faceplates and concrete fill, and the diaphragm plates act as shear reinforcement besides holding the system together. The design of the structures serving as the containment pressure boundary is performed in accordance with the provisions of ASME Boiler and Pressure Vessel Code (BPVC) as described in NEDC-33926P (Reference 3.5-5). The Steel-plate Composite Containment Vessel (SCCV) is designed in accordance with NEDC-33926P, as described in Subsection 3.5.3.1.

Similarly, the Class MC containment metal components are designed in accordance with the provisions of ASME BPVC, Section III, Division 1, Subsection NE (Reference 3.5-6).

ANSI/AISC N690 (Reference 3.5-7) that has been endorsed by U.S. NRC RG 1.243 (Reference 3.5-8), along with NEDC-33926P provide the specifications for the design, fabrication, construction, examination, and inspection of RB Steel Bricks™ and steel structures that do not provide the containment pressure boundary and for the containment internal structures.

These U.S. codes and standards are adopted for the BWRX-300 steel-plate composite structures (Steel Bricks™) since there are no equivalent standards or regulatory guidance in Canada.

Clause 6.1.2 of CSA N291 (Reference 3.5-9) permits the use of alternate design methods for design of nuclear structures and concrete containments in Canada. Requirements for design, fabrication, construction, examination, and testing of containment, containment internal structures, RB, and their foundations presented in Subsections 3.5.2 through 3.5.5 ensure compliance to the regulatory requirements in CNSC REGDOC-2.5.2 and meet the intent and ensure a level of safety and performance commensurate with the applicable Canadian standards.

3.5.1.1 Structural Analysis Criteria for Seismic Category A Structures

In accordance with requirements in CNSC REGDOC-1.1.2, Section 4.5.5 and CNSC REGDOC-2.5.2, Sections 7.13.1, 7.15.1, 7.22 and 8.6, the RB, containment and the containment internal structures are analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine structural design demands resulting from various design loads and design load combinations. Evidence of qualification of these computer programs, including a description of the programs and extent of use, is presented in Appendix 3B.

The following Finite Element (FE) analyses are performed to obtain stress demands for the design of the BWRX-300 RB, containment, and containment internal structures:

- 1-g static SSI analyses
- Static and quasi-static analyses
- Thermal stress analyses
- Seismic SSI analyses

Static analyses provide design demands on the RB integrated structures from dead loads, live loads, earth pressure loads, hydrostatic and hydrodynamic loads, severe and extreme environmental loads, plant operating loads during normal operation, testing and abnormal plant conditions. Thermal analyses provide stress demands due to normal operating and accidental load conditions. Design Basis Earthquake (DBE) seismic demands are obtained directly from the results of one-step approach SSI seismic analyses discussed in Subsection 3.3.1.2.

The effect of interaction with the surrounding subgrade is incorporated in the analyses of the deeply embedded integrated RB by considering the surrounding soil and rock as a layered half-space continuum. The geotechnical design parameters used as input for the static and thermal analyses are developed as described in Subsection 3.5.2.2.

3.5.1.1.1 FE Model of Integrated RB Structure

To determine internal forces resulting from various loads and loading combinations, a detailed structural model is developed for the integrated RB, containment, and containment internal structures, including their foundations, penetrations, and openings, following the general FE modeling guidelines for the integrated RB structure discussed in Subsection 3.3.1.2 and NEDO-33914 Revision 2 (Reference 3.5-10), Section 5.1.1. The integrated structural FE model adequately represents the RB structural configuration for all main structural members and meets the mesh refinement and quality attributes required for calculation of structural stress demands. The use of the common model enables the FE results obtained from the different analyses to be directly combined in design load combinations per governing design codes.

Materials properties assigned to the integrated RB model depend on the analyzed loads and resulting stress responses. Unit weight properties are assigned to the models used for the 1-g static SSI analyses to adequately simulate gravity and earth pressure loads. The dynamic model of the integrated RB used for the seismic SSI analyses is assigned seismic mass inertia properties as discussed in Subsection 3.3.1.2.

As discussed in Subsection 3.3.1.2, stiffness properties are assigned to the SCCV and RB to reflect effective stiffness for load combinations without accidental thermal load. For load combinations with accidental thermal load, reduced stiffness is considered to account for the cracking effects on the redistribution of forces and moments. Spring elements are also used in the integrated FE element model to represent the stiffness of the connections between the different structural members that are designed to relief 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 sub-structuring method as the seismic SSI analyses described in Subsection 3.3.1.2. LB equivalent linear stiffness properties and UB unit weight properties assigned to the subgrade model used in the analyses are discussed in Subsection 3.5.2.2.

Maximum dynamic responses of the SSI system that are equivalent to its static response under 1-g gravity load are calculated by applying on the 1-g SSI analyses model an equivalent static 1-g excitation in the vertical direction as vertically propagating compression wave. To simulate 1-g excitation, a harmonic acceleration time history is used with:

- A low frequency equal to the analysis frequency increment, and
- An amplitude equal to the Earth's gravity (g).

The 1-g excitation is applied at control point located at the surface of the site free-field model.

Stress demands obtained from the one-step 1-g static SSI analyses include the effects of static earth pressures simulated by the interaction of the integrated RB structural model with the subgrade FE model. Shell elements at the surface of the subgrade are included in the SSI model to simulate the applicable overburden inertia loads from the surrounding Power Block foundations and other surcharge loads.

Contact springs are used at the interfaces of the RB structure with the surrounding subgrade as discussed in Subsection 3.3.1.2. In accordance with the FE modeling guidance in NEDO-33914 Revision 2, Section 5.1.1, the following stiffness properties are assigned to the contact springs in the models used for the 1-g static SSI analyses to provide UB lateral soil pressures on the RB below grade exterior walls:

1. The contact springs in the direction normal to the RB exterior walls are assigned properties representing UB stiffness conditions at the SSI interfaces.
2. The friction at the RB exterior walls is not considered by assigning very low stiffness properties to the contact springs in vertical and tangential direction.

Results obtained from these contact spring elements serve for calculation of earth pressures on the below grade RB shaft exterior wall and mat foundation.

Subgrade Modeling Assumptions for Deeply Embedded RB

Per NEDO-3914, Section 5.1.2, the following assumptions related to the modeling of the subgrade are introduced in the 1-g Static SSI analyses to enable an efficient calculation of stress demands on the RB structure due to pressure loads from soil and rock surrounding and supporting the RB shaft:

1. The properties of the subgrade materials are represented by linear elastic constitutive models
2. The non-linearities at soil-structure interfaces are not considered
3. The rock mass is assumed continuous and the presence of cavities, fracture zones, joints, bedding planes, discontinuities and other weak zones is not considered

The soil and rock strata in the 1-g static SSI models are modeled based on the principles of continuum mechanics using isotropic linear elastic properties. Possible fracture zones, joints, bedding planes, discontinuities and cavities in the rock are not explicitly included in the design SSI analyses models. Bounding properties assigned to the soil and rock materials are discussed in Subsection 3.5.2.2.

The effects of non-linearities at soil-structure interfaces are addressed by using elastic contact spring stiffness properties that provide bounding structural demands.

Rock with disadvantageous fracture zones, joints, bedding planes and discontinuities is reinforced to create a more self-supporting rock mass. If needed, rock reinforcements are provided as initial ground support. The rock reinforcements and any other support provided during the excavation and construction may degrade and is inaccessible after construction. Therefore, the design addresses the rock loads remaining after the initial ground support degrades by including the potential weight of the rock in the static 1-g SSI analysis or by applying additional pressures on the RB outer shaft wall. Additional horizontal pressure loads are also applied on the model to account for possible residual stresses in the DNNP rock mass.

RB Design Earth Pressure Load Validation

Validations of the earth pressure loads are to be performed following the guidelines in Section 5.1.3 of NEDO-33914 Revision 2 to ensure the 1-g SSI static analysis provides conservative earth pressure design demands on the deeply embedded RB structure.

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.13.1 and NEDO-33914 Revision 2, Section 4, Foundation Interface Analyses (FIA) are performed on models representative of the non-linear constitutive behavior of soil and rock materials surrounding the RB shaft and employ non-linear interface modeling features capable of capturing the effects of non-linearities at the subgrade structure contact surfaces. The results of the FIA are to be used for validation of the design earth pressures following the guidance of Section 5.1.3 of NEDO-33914 Revision 2.

3.5.1.1.3 Static and Quasi-Static Load Analyses

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1, the following static and quasi-static analyses are performed on the integrated RB FE model to calculate structural stress demands due to:

- Live loads
- Crane loads
- Structural Integrity Test (SIT) and accident condition containment internal pressure load including differential containment and RB sub-compartment loads
- Horizontal hydrostatic pressure loads on pool walls
- Groundwater pressure loads on the integrated RB common mat foundation and below-ground exterior wall
- Extreme wind and tornado loads on RB roof and exterior wall
- Rain and snow loads
- Seismic water sloshing and breathing mode quasi-static pressure loads on pool walls
- Quasi-static pressure High Energy Line Break (HELB) loads (jet impingement, blast loads)
- Equipment and pipe reaction loads including RPV reaction loads.
- Post-accident internal flooding loads

The analyses of global static and quasi-static loads that can affect the global response of the integrated RB consider the effect of subgrade stiffness. Following the sub-structuring methodology, design demands from these loads are obtained from subgrade stiffness impedance analyses performed on models consisting of two parts:

- Super-element representing LB stiffness of the subgrade surrounding the RB, and

- Integrated FE model of the RB, containment and containment internal structures described in Subsection 3.5.1.1.1.

The super-elements define the stiffness of the subgrade at the nodes of the RB interfaces with the surrounding soil. The stiffness properties of the super-elements are developed using a layered 3-D solid FE model. Subgrade stiffness properties assigned to the super-elements are described in Subsection 3.5.2.2. To adequately simulate half-space boundary conditions, the depth of these models is deeper than three times the largest foundation dimension. The horizontal extent of these models is more than three times the RB shaft diameter.

The nodes of the super-element are coincident with the nodes of the integrated RB FE structural model. The coincident super-element and structural model nodes are connected by contact spring elements as described in Subsection 3.5.1.1.2. LB stiffness properties are assigned to these contact spring elements to yield larger structural deformations and conservative design stress demands. Equivalent linear subgrade stiffness properties assigned for the subgrade stiffness impedance static analyses are discussed in Subsection 3.5.2.2.

Fixed bases analyses are performed for the local loads with smaller magnitudes that do not affect the Integrated RB mat common mat foundation or global response.

Demands due to hydrostatic lateral pressure loads are obtained from static analyses of the integrated RB model with vertical supports applied to all mat foundation nodes. Demands from the upward buoyant pressures on the mat foundation are obtained from a static analysis of the integrated RB structural model with vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat. The results from the two groundwater load analyses are enveloped and then combined with the results of the 1-g SSI analysis cases to obtain earth pressure and groundwater load demands for the design of integrated RB structure.

Additional Rock Pressure load analyses are performed to account for possible residual horizontal stresses in the DNNP rock strata. Two boundary conditions are considered for these analyses that result in conservative stress demands:

1. Vertical supports established at all mat foundation nodes and horizontal supports established at the central node of the mat; and
2. Vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat.

The results of these two sets of additional static rock pressures analyses are enveloped and then combined with the results of the 1-g SSI analyses to ensure the RB structural design adequately addresses the effects of anisotropic and heterogenous rock behavior and accounts for potentially unstable rock mass loads.

3.5.1.1.4 Thermal Stress Analyses

To calculate structural stress demands due to the normal operating and DBA temperature loads, sub-structuring thermal stress analyses are performed on the integrated RB FE structural model coupled with super-element representing UB stiffness of the subgrade.

Stiffness properties are assigned to the Steel Bricks™ shell elements to account for the stiffness reduction effects under normal operating and DBA temperature loads. The corresponding structural stiffness conditions are used for the analyses for design loads that occur in combination with the normal and accident thermal loads.

For the thermal analyses, UB stiffness properties are assigned to the super-element modeling the subgrade and to the contact elements modeling the soil-structure interfaces resulting in conservative thermal stress demands for the design of the RB and containment structures. Equivalent linear subgrade stiffness properties assigned for the thermal stress analyses are discussed in Subsection 3.5.2.2.

3.5.2 Foundations

This section presents general design rules for the common Steel Bricks™ mat foundation supporting the integrated RB structure. Design rules for other foundations are discussed in Chapter 9B, Section 9B.3.

3.5.2.1 Applicable Codes, Standards and Other Specifications

Applicable codes, standards and specifications for the containment and RB common Steel Bricks™ foundation are the same as those for the superstructures.

The jurisdictional boundary for the application of the NEDC-33926P to the containment is the portion within the perimeter or exterior surface of the SCCV as shown in Figure 3.5-1.

The jurisdictional boundary for application of the ANSI/AISC N690 to the non-pressure retaining portion of the common foundation is the portion spanning from the exterior surface of the SCCV to the exterior surface of the RB (See Figure 3.5-1).

3.5.2.2 Bounding Subgrade Design Parameters

Bounding subgrade parameters are determined based on data available prior to the completion of the complete characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative subgrade property inputs adequately address uncertainties related to the use of incomplete characterizations of the DNNP site geotechnical and seismic conditions.

Based on the information from the available groundwater flow patterns and conditions at the DNNP site provided in NK054-REP-01210-00011 (Reference 3.5-11) and NK054-REP-07730-00005 (Reference 3.5-12), an Upper Bound groundwater level at elevation 85 m CGD corresponding to a depth of 3 m below the plant grade at elevation 88 m CGD is considered a parameter for the bounding design.

The geotechnical and hydrological investigations of the DNNP site have been completed and bounding subgrade design parameters determined (see Chapter 2, Subsection 2.7.5). The data collected from ground water measuring wells at the DNNP site indicate an upper bound nominal water table at a shallower depth of 2 m. The increase of an additional meter in the nominal ground water table elevation results in a 6% higher magnitude of the total force from ground water pressure load than the one calculated using the bounding design ground water table at 3 m depth.

The exterior RB wall is the main structural member resisting the below grade lateral pressures applied on the RB integrated structures. These below grade lateral loads include the static earth pressure, ground water hydrostatic pressure, and additional rock pressure that account for a large majority of the demand on the below grade portion of the exterior RB wall in approximately equal shares. Therefore, the effect of the marginal 6% increase in the ground water pressure, that represents no more than a third of the total structural demand on the exterior RB, is negligible and well bounded by the available structural design margins (see Chapter 9B, Appendix 9B.G).

Identification and evaluation of potentially liquefiable cohesionless soil strata under the BWRX-300 Power Block structures is performed in accordance with CSA N289.3 (Reference 3.5-13) and in compliance with requirements of CNSC REGDOC-2.5.2, Section 7.15.1.

3.5.2.2.1 Bounding Equivalent Linear Subgrade Static Profiles

As described in Subsection 3.5.1.1, the structural design demands due to static earth pressures on the RB below grade exterior walls are obtained from the 1-g static analyses of the integrated RB FE model embedded in a layered half-space continuum model representing the surrounding soil and rock. To account for the interaction of the RB integrated structures with the surrounding subgrade, super-elements representing the stiffness properties of the layered subgrade materials are used in the static and thermal analyses, as described in Subsection 3.5.1.1.

The 1-g static SSI analyses, subgrade impedance analyses and thermal stress analyses use profiles of bounding equivalent linear soil and rock properties developed using information form 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_o) are determined using effective angle of friction (ϕ_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 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 ν_{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 ν_{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 ν_{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 (ν_{st}) are calculated as follows using the at-rest lateral (k_0) coefficient values provided in Table 3.5-1:

$$\upsilon_{st} = \frac{K_0}{1 + K_0}$$

LB soil and rock stiffness properties are used for the static analyses including the 1-g SSI analyses resulting in larger deformation at soil-structure interfaces and conservative design stress demands. Thermal stress analyses are performed using UB soil and rock stiffness properties resulting in conservative thermal stress demands.

3.5.2.2.2 Soil Bearing Stability

The stability of soil supporting the BWRX-300 structural foundations is demonstrated in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2 and per the regulatory guidance of US NUREG-0800 (Reference 3.5-20), SRP 2.5.4.10, and IAEA Safety Guide No. NS-G-3.6 (Reference 3.5-21).

The bearing capacity of the rock supporting the RB mat foundation is discussed in Chapter 2, Subsection 2.7.3.3.

Since the RB is deeply embedded, the bearing surface of the common foundation is below the depth of frost action to meet the requirements of NBC (Reference 3.5-22), Article 4.2.4.4.

Chapter 2, Subsection 2.7.3.3 also discusses the bearing capacity of the component in-situ soil materials supporting the shallow foundations surrounding the RB.

The calculation of the dynamic bearing pressure demands under DBE loads from the results of the seismic SSI analyses is described in Subsection 3.3.1.2.

Per Article 4.35 of IAEA Safety Guide No. NS-G-3.6, safety factors against potential bearing capacity failure of the subsurface materials depend on the method of bearing capacity evaluation and site conditions. If a conventional bearing capacity method is used, safety factors are not less than 3 under static loads and 1.5 under loads that include DBE.

3.5.2.2.3 Foundation Stability

Foundation stability is assessed against sliding and overturning due to earthquakes, wind and tornados, and flotation in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2, following the regulatory guidance of US NUREG-0800, SRP 3.8.5 and in accordance with Clause 5.9 of CSA N289.3.

Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB since, in accordance with Sections 7.2.1 and 7.2.2 of ASCE/SEI 43 (Reference 3.5-23), its centre of gravity is below the grade elevation, and the structure is inherently stable against sliding and overturning. The foundation stability of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay that are supported by surface mounted foundations is checked to ensure that there is no adverse interaction with the Seismic Category A RB during a DBE level event. Stability of the surface mounted foundations surrounding the RB under DBE loads is evaluated using the results of the seismic SSI analyses as described in Subsection 3.3.1.2.

Safety factors against sliding and overturning under normal operating conditions that include unfactored combination of dead loads, soil pressure loads, and design wind, and accidental conditions that include combination of dead loads, soil pressure loads, and DBE loads are presented in Table 3.5-3.

3.5.2.3 Loads and Load Combinations

3.5.2.3.1 Design Loads

Design loads of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For foundation stability against flotation, the site-specific design basis flood is considered.

3.5.2.3.2 Design Load Combinations

Design load combinations of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For the stability against flotation of the integrated RB foundation, the load combination is in accordance with U.S. NUREG-0800, SRP 3.8.5, where the design basis flood is considered in combination with the dead load.

3.5.2.4 Design and Analysis Procedures

The design of the deeply embedded foundation and foundation stability evaluations are in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1 and follow the BWRX-300 specific criteria and guidelines in NEDO-33914 Revision 2.

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 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, Lo) 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

- 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 DECs.

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 DECs 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.

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 DECs.

Evaluation Criteria for Structure Interaction Under Seismic and Extreme Wind

The interaction of the RB structure with the adjacent RWB, CB, TB and Reactor Auxiliary Bay is discussed in Subsections 3.3.1.2 and 3.3.2.8.

The stability of foundations under DBE and design basis tornado wind loads are checked following the criteria in Subsection 3.5.2.2.

RB Design for Impulsive and Impactive Loads

The RB structure is designed for impulsive and impactive loads per the requirements of Sections 7.15.1 and 7.15.3 of CNSC REGDOC-2.5.2 and the regulatory guidelines of U.S. NUREG-0800, SRP 3.8.4.

The RB design for impulsive and impactive loads follows the provisions of ANSI/AISC N690 and the relevant regulatory guidance of U.S. NRC RG 1.243.

Criteria used to define the heavy loads considered in the RB design are described in Subsection 3.4.5.1.

3.5.5.4.2 Robustness Acceptance Criteria for RB Structure

Refer to Subsection 3.5.6.1 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 RB structure, which satisfy the requirements in CNSC REGDOC-2.5.2, Section 7.22.3.

3.5.5.5 Materials, Quality Control and Special Construction Techniques

3.5.5.5.1 Materials

Materials used in construction of the RB structure outside of the containment are in accordance with ANSI/AISC N690, Section NA3.

Details of materials used in the construction of the RB are provided in Chapter 9B, Subsection 9B.2.3.4.

3.5.5.5.2 Quality Control

Quality control procedures are established and implemented during the construction and inspection phases of the RB structure. These procedures cover the fabrication, furnishing, and installation of each structural item in the RB and specify the inspection and documentation requirements in accordance with the requirements in ANSI/AISC N690, Section NA5, Chapter NN with supplemental guidance provided in U.S. NRC RG 1.243.

3.5.5.5.3 Special Construction Techniques

The BWRX-300 Seismic Category A structures at the DNNP site are built using a modular construction technique using Steel Bricks™. (see Section 3.5.1).

The quality control procedures used in the structural modularization process implemented in the construction of the Steel Bricks are outlined in Subsection 3.5.5.5.2. These procedures are employed at the fabrication shop and the construction-site (both outside and inside the deep excavation pit necessary for the construction of RB), including pre-fabrication and pre-assembly, to ensure the Steel Bricks™ modular assemblies meet the necessary material quality, fabrication, and installation requirements per the applicable code of records.

For the preferred method of construction for the deeply embedded BWRX-300 RB shaft, refer to Section 1.4 of NEDO-33914 Revision 2.

For plant construction and commissioning activities, refer to Chapter 14.

3.5.5.6 Testing and In-Service Inspection Requirements

Per CNSC REGDOC-2.5.2, Section 7.15.2, periodic inspection, and in-service monitoring programs are implemented to ensure the RB structure continues to meet its functional and performance requirements.

Sections 3.2 through 3.4 of NEDO-33914 Revision 2 describe the approaches and guidelines for the BWRX-300 in-service testing, monitoring, and monitoring programs.

NEDC-33926P describes the in-service inspection and testing guidelines for the Steel Bricks™ to ensure that the integrated RB structures satisfy their functional and performance requirements through all phases of the plant's life cycle. The BWRX-300 implements a Structures Monitoring and Aging Management Program (SMAMP) that monitors the condition of structures and manages aging effects in accordance with CSA N291, clauses 9 and 10 and in compliance with CNSC REGDOC-2.5.2, Section 7.17. The program demonstrates that the facility is constructed to the requirements in the design drawings and specifications. A research and development program is also established to demonstrate the adequacy of Steel Bricks™ to maintain the structural integrity of the integrated RB structures and of inspection methods used in compliance with CNSC REGDOC-2.5.2, Section 5.4.

3.5.6 Robustness Design of Seismic Category A Structures

Consistent with the Level Four D-in-D requirements discussed in Subsection 3.1.6 and in Section 6.1 of CNSC REGDOC-2.5.2, the BWRX-300 containment and RB are robust structures, tolerant of a large spectrum of faults with a gradual degradation in their effectiveness, that would not fail catastrophically under operational states, DBAs and DECs.

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, DECs 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 DECs and evaluate the consequences of the DECs.

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 DECs.

A reasonable level of survivability of the structure under postulated DECs 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 DECs.

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 DECs.

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 DECs.

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

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 (≥ 95%) of Low Probability (≤ 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

- 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."

3.5-2 CNSC Regulatory Document REGDOC-1.1.5, “Reactor Facilities: Supplemental Information for Small Modular Reactor Proponents.”

3.5-3 CNSC Regulatory Document REGDOC-3.5.3, “CNSC Processes and Practices, Regulatory Framework.”

3.5-4 CNSC Regulatory Document REGDOC-2.5.2, “Design of Reactor Facilities: Nuclear Power Plants.”

3.5-5 NEDC-33926P, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.

3.5-6 ASME BPVC-III NE-2021, “BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components,” American Society of Mechanical Engineers.

3.5-7 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American Institute of Steel Construction.

3.5-8 USNRC Regulatory Guide 1.243, “Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments.”

3.5-9 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.

3.5-10 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC. (Reference 2.7-35),

3.5-11 NK054-REP-01210-00011 R001, “Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects,” Ontario Power Generation. 2009 (Reference 2.7-1)

3.5-12 NK054-REP-07730-00005 Rev. R000, Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment,” Ontario Power Generation. 2009 (Reference 2.7-41)

3.5-13 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.

3.5-14 NK054-REP-01210-0418696, “Geologic and Geophysical Evaluation, Darlington Site Investigation – Phase III (Field Work), AMEC Report No. D0053/RP/002 R01, Volumes 1 and 2,” Ontario Power Generation. 2012 (Reference 2.7-36)

3.5-15 NK054-REP-01210-00098 R000, Geotechnical Data Report – R2, Darlington New Nuclear Project Geotechnical Investigation, EXP Services Inc. Project No. BRM-00025482-A0,” Ontario Power Generation. 2013 (Reference 2.7-37).

3.5-16 DM 7.01, “Soil Mechanics,” Naval Facilities Engineering Command. 1986 (Reference 2.7-38),

3.5-17 EPRI EL-6800, “Manual on Estimating Soil Properties for Foundation Design,” Electric Power Research Institute. 1990 (Reference 2.7-39)

3.5-18 Carter, M. and Bentley, S., “Soil Properties and their Correlations,” John Wiley & Sons, West Sussex, UK, 2016. (Reference 2.7-40)

3.5-19 Lo, K.Y., and B. Lukajic, “Predicted and Measured Stresses and Displacements around the Darlington Intake Tunnel,” Canadian Geotechnical Journal, 21:147-165. (Reference 2.7-33)

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.

3.5-40 CSA S16, "Design of Steel Structures," CSA Group.

3.5-41 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group

3.5-42 USNRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."

3.5-43 NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Nuclear Energy Institute.

3.5-44 ASME BPVC-II D, "Section II: Materials-Part D-Properties (Customary), American Society of Mechanical Engineers.

3.5-45 ASME BPVC-III, "Appendix XXVII: Design by Analysis for Service Level D," American Society of Mechanical Engineers.

Table 3.5-1: As-Built Static Properties for Soil Layers

Layer	Layer Thick. (m)	Total Unit Weight (kN/m^3)	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

Table 3.5-2: Summary of Static Rock Properties

Layer	Total Unit Weight (kN/m^3)	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

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*
where			
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	$0.67F_y$

Table 3.5-5: Acceptance Criteria for Containment Closure Head

Service Level	Acceptance Criteria[*1]			
	P_m	P_L	P_L+P_b [*2]	P_L+P_b+Q
Test Condition	0.8 S_y	1.15S_y	1.15S_y	N/A[*3]
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A[*3]
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[*4] 1.0 S_y	1.8 S_{mc} or[*4] 1.5S_y	1.8 S_{mc} or[*4] 1.5S_y	N/A[*3]
Level D	S_f	1.5S_f	1.5S_f	N/A[*3]

*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.

Table 3.5-6: Acceptance Criteria for Other MC Components

Service Level	Acceptance Criteria*1			
	P_m	P_L	P_L+P_b *2	P_L+P_b+Q
Test Condition	0.8 S_y	1.15S_y	1.15S_y	N/A*3
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A*3
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*4 1.0 S_y	1.8 S_{mc} or*4 1.5S_y	1.8 S_{mc} or*4 1.5S_y	N/A*3
Level D	S_f	1.5S_f	1.5S_f	N/A*3
Post-flooding Condition	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5S_y	1.8 S_{mc} or*4 1.5S_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

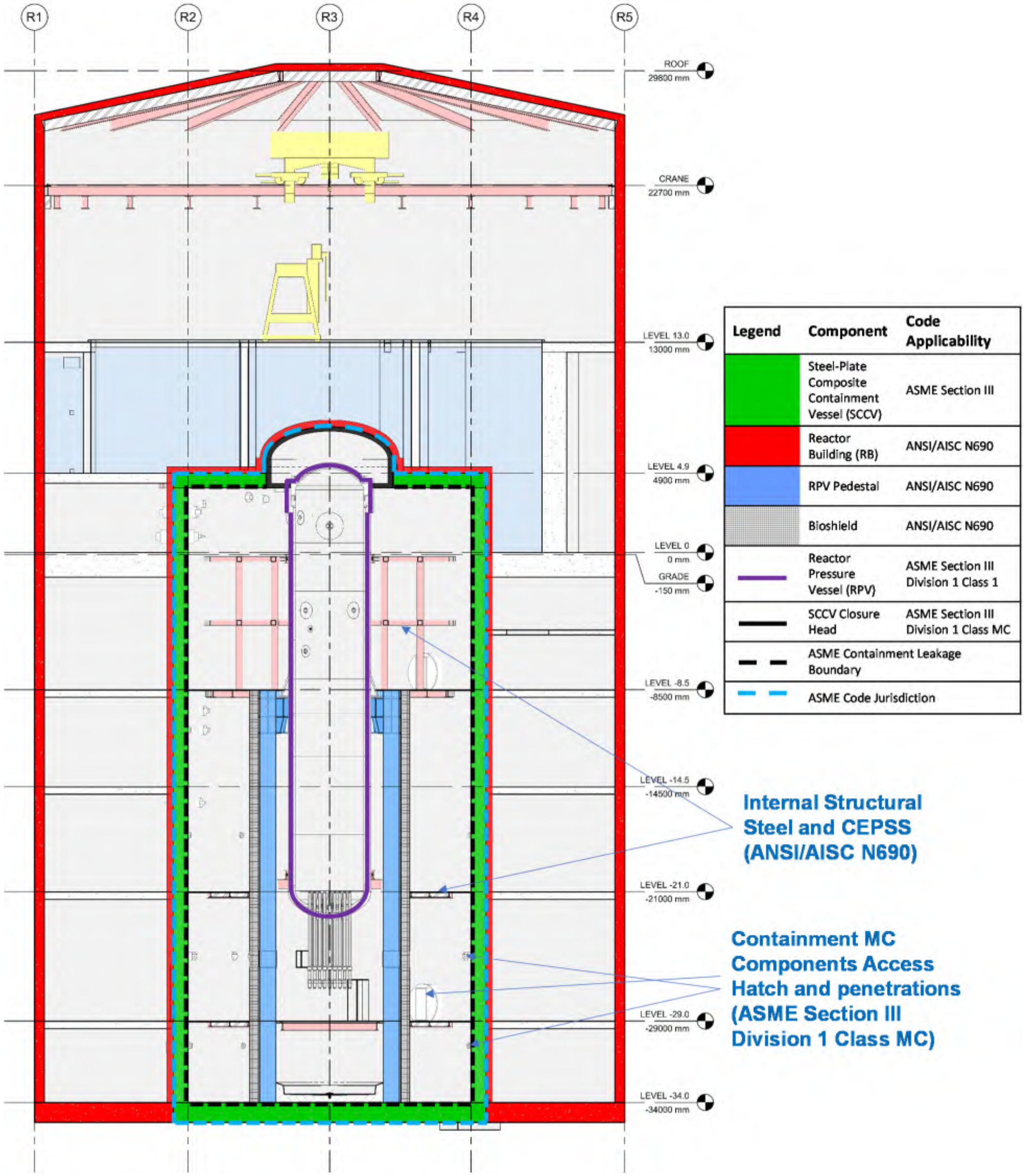


Figure 3.5-1: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

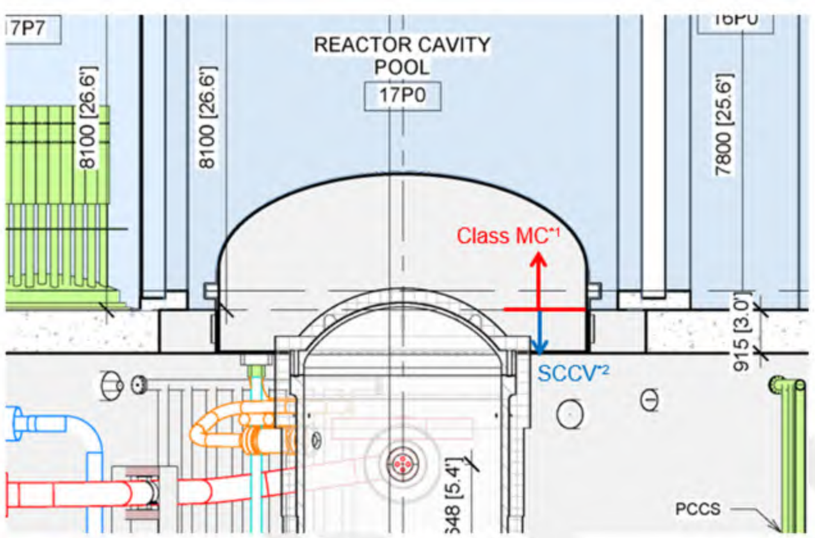


Figure 3.5-2: Containment Closure Head Structure Boundary

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

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

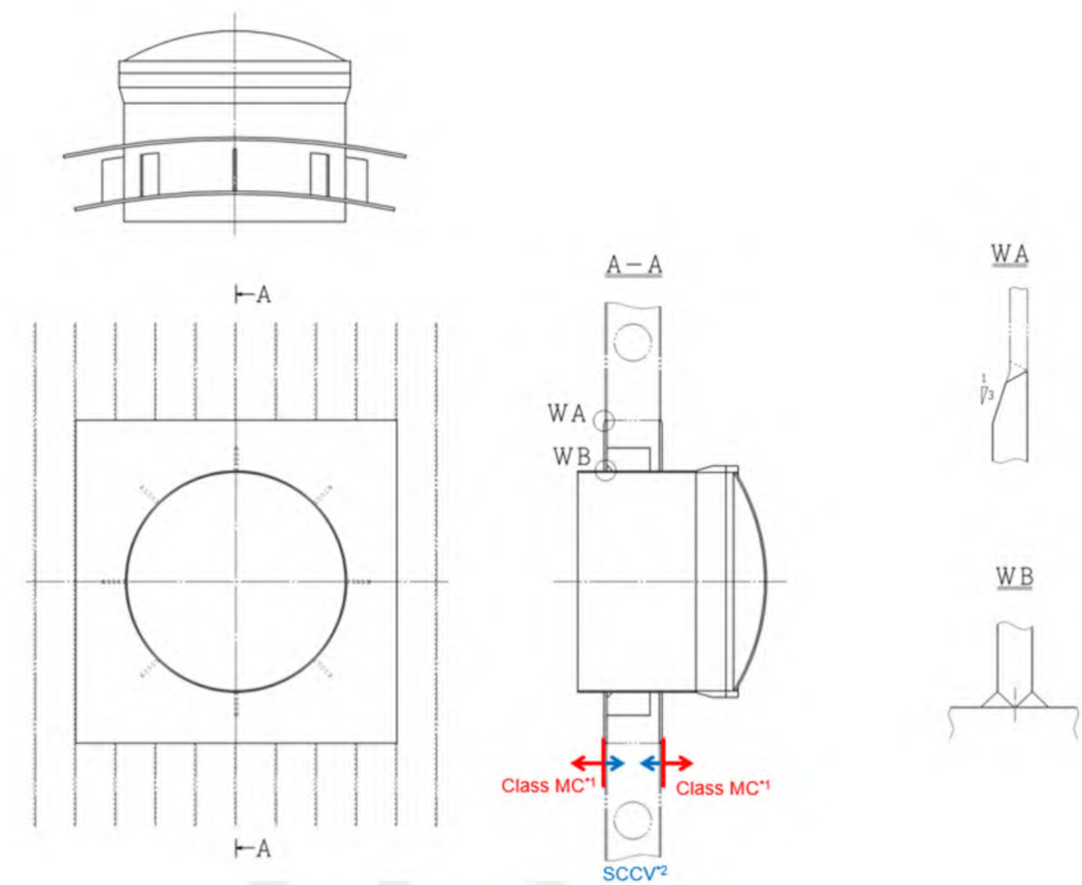


Figure 3.5-3: Access Hatch Code Jurisdictional Boundary

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

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

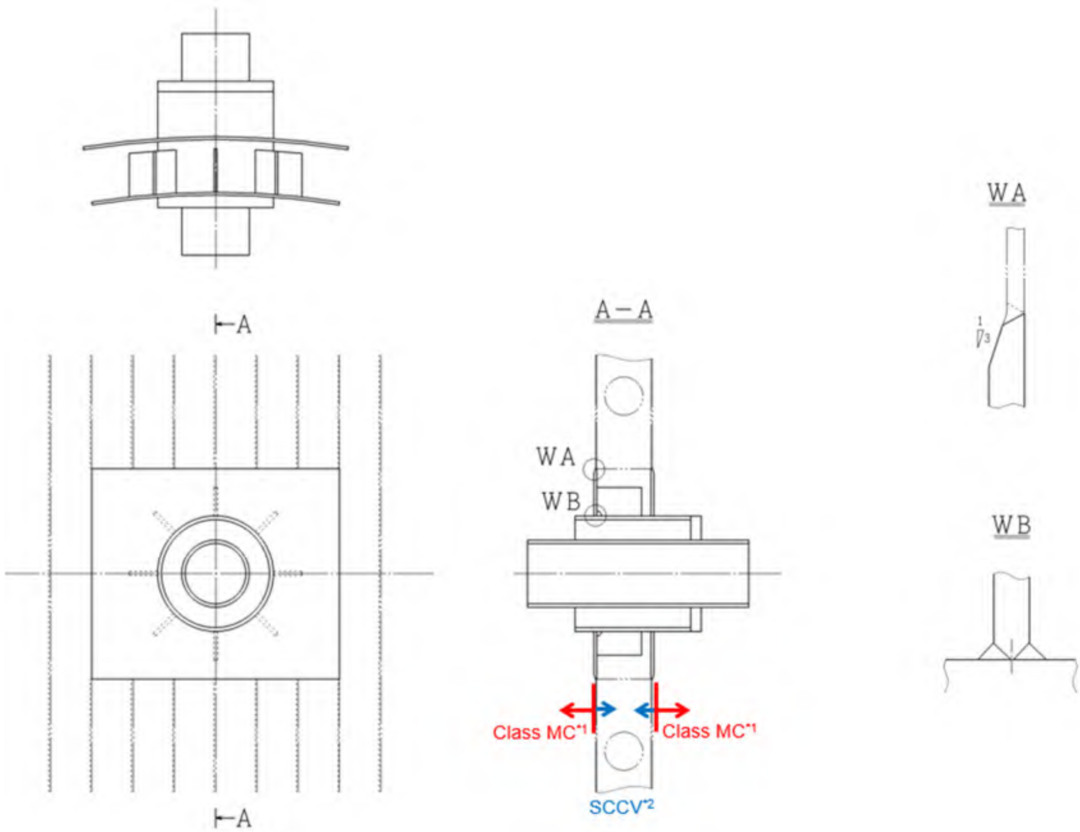


Figure 3.5-4: Penetrations Jurisdictional Boundary

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

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

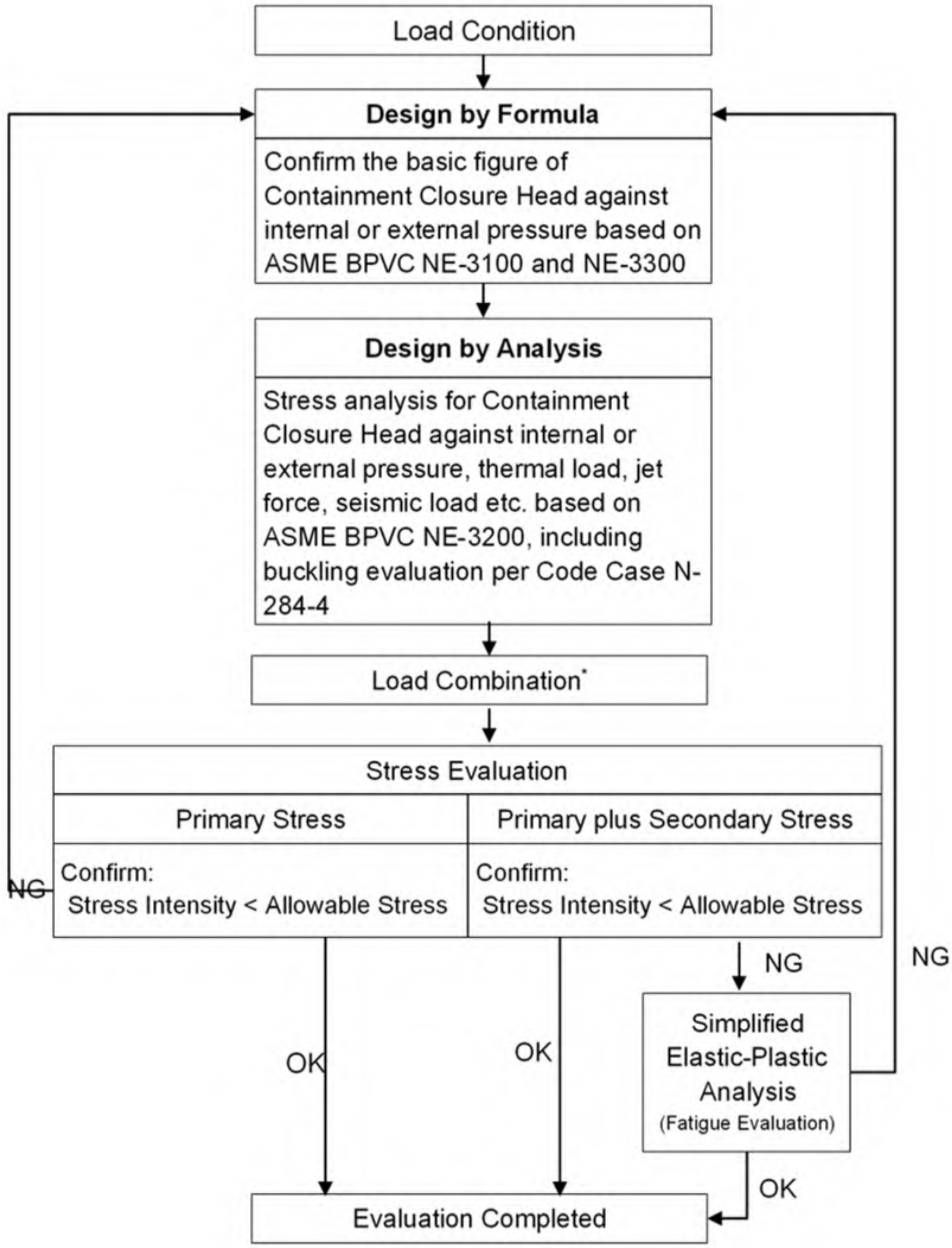


Figure 3.5-5: Design Procedures for the Containment Closure Head

*: Steel Portion: U.S. NRC RG 1.57 and U.S. NUREG-0800 SRP 3.8.2

Concrete Portion: NEDC-33926P

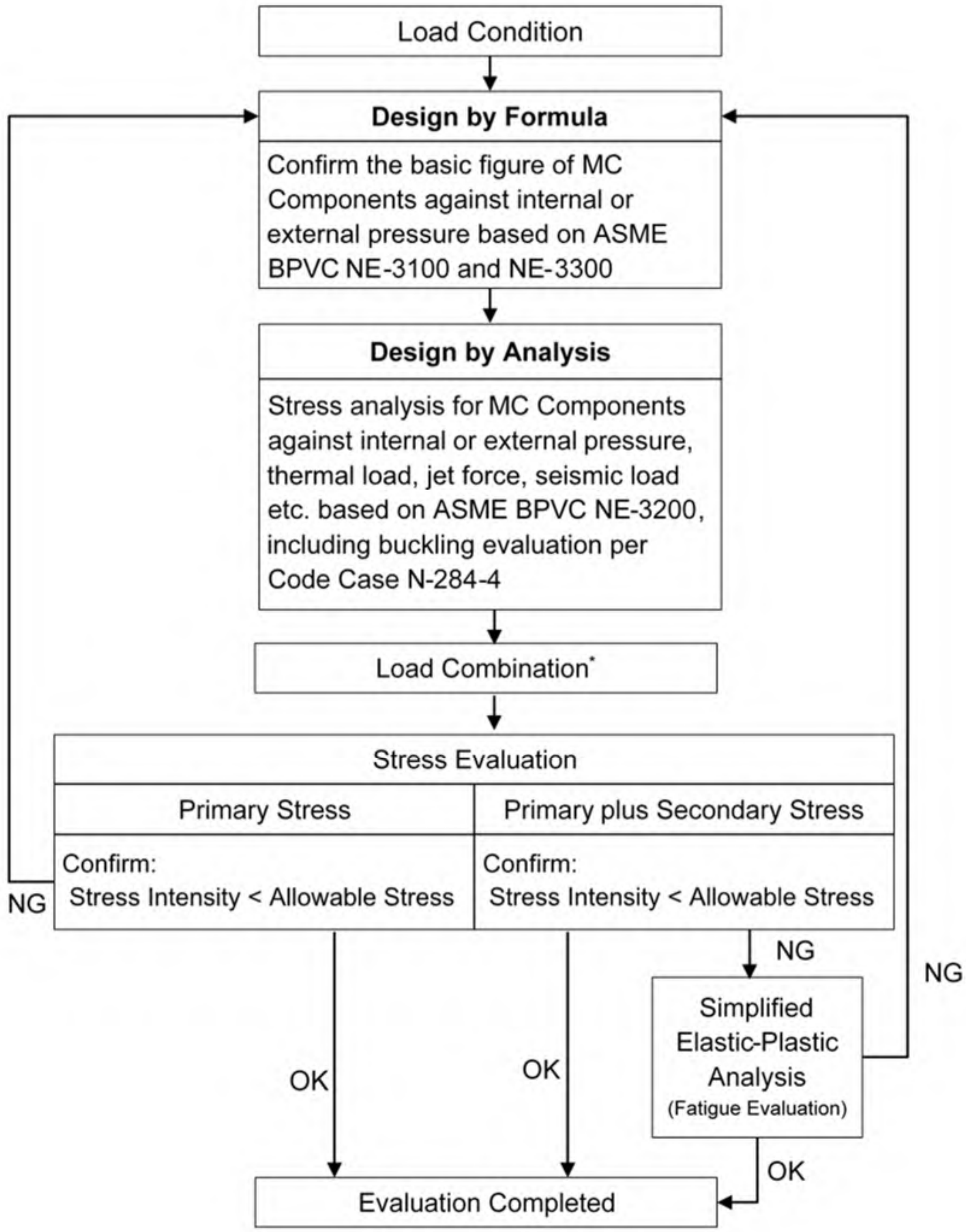


Figure 3.5-6: Design Procedures for the MC Components

*: 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)

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 DECs.

3.6.3.2.5 Seismic Events

Seismic design parameters and associated seismic events defined in Subsection 3.3.1 are used in qualification of mechanical system components. The magnitude of seismic events is determined by Ground Response Spectra accelerations applied to Building Structures and creating Amplified Response Spectra (ARS) accelerations at various building elevations where the components are located. These ARS accelerations are used in qualification of Mechanical systems and equipment and a determination of component and system structural and/or functional capacity is determined. Seismic Category A (passive components) require only structural code adequacy. Seismic Category B (active components) such as valves and pumps require both structural code adequacy and functional capacity under seismic demand. Chapter 3, Subsection 3.9.3 provides seismic qualification methodology to assure both component structural and/or functional capacity under seismic operational conditions are met.

The seismic categorization of SSC is defined in Section 3.2 and related to the seismic category to the more general safety strategy defense lines. In summary, Defense Lines 3 and 4b are generally Seismic A or B and Defense Line 4b also has an additional requirement of satisfying the plant-level High Confidence of Low Probability of Failure (HCLPF) criteria.

3.6.3.2.6 Non-LOCA Fault

Non-LOCA Fault consists of any DEC event not considering a LOCA which has a significantly low frequency of occurrence to be considered as a faulted event.

3.6.3.2.7 Plant Testing

Plant testing events are occasional operating loads imposed during pre-operational testing or periodic operational testing.

3.6.3.3 Classification of Components

All SSC of the BWRX-300 design are designated by Safety Class, Quality Group, and Seismic Category according to guidance in Section 3.2 which are consistent with their Defence-in-Depth categorization defined in the BWRX-300 Safety Strategy, in Section 3.1. Appendix 3A provides the Classification Table for Plant SSC.

3.6.3.4 Establishment of Design, Service, and Test Loadings and Limits

Design, Service, and Test Loadings and Limits for fixed equipment components and supports are in accordance with ASME BPVC Section III Division 1 (Reference 3.6-5).

For IEEE Equipment, SC1 electrical equipment is evaluated with respect to the load combinations in this document using IEC/IEEE 60980--323 and IEC/IEEE 60980--344 Acceptance Criteria, Codes and Standard (References 3.6-18 and 3.6-19).

For SC1, actuators and power operated valve assemblies are evaluated with respect to the load combinations in this document in accordance with the provisions of ASME Standard QME-1.

3.6.3.5 Acceptance Criteria

Components and supports comply with the design rules established for design, service, and test loadings in the appropriate with the appropriate subsection of the ASME BPVC, Section III, Division 1 (References 3.6-1 through 3.6-8).

Design documentation is completed in accordance with the requirements of the Subsection of the ASME BPVC applicable to the component or support.

3.6.3.6 Loading Criteria

3.6.3.6.1 Loading Conditions

The loadings that are considered in designing a component include, but are not limited to, those in (a) through (g) below:

a. Internal and external pressure
b. Impact loads, including rapidly fluctuating pressures
c. Weight of the component and normal contents under operating or test conditions
d. Superimposed loads such as other components, operating equipment, insulation, corrosion resistant or erosion resistant linings, and piping
e. Wind loads, snow loads, vibrations, and earthquake loads, where specified
f. Reactions of supporting lugs, rings, saddles, or other types of supports
g. Temperature effects

As appropriate ASME BPVC, Division 1, Section III, Paragraph, NB-3111, NCD-3111, NE-3111, NF-3111 or NG-3111, is applied for a complete list of required load conditions to consider.

Consistent with the ASME BPVC Section III Division 1, the stresses resulting from differential anchor movements during dynamic events are considered secondary stresses.

3.6.3.6.2 Design Loadings

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3112, NCD-3112, NE-3112, NF-3112 or NG-3112, as applicable.

3.6.3.6.3 Service Conditions

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3113, NCD-3113, NE-3113, NF-3113 or NG-3113, as applicable.

Each service condition to which the components may be subjected is classified in accordance with Service Limits designated in the Component Design Specifications in such detail as will provide a complete basis for design, construction, and inspection.

For ASME BPVC Section III Division 1, Class 1 Components, the requirements of (1) and (2) below apply.

1. Level B Conditions. The estimated duration of service conditions for which Level B Limits are specified are included in the Design Specifications.
2. Level C Conditions. The total number of postulated occurrences for all specified service conditions for which Level C Limits are specified are limited to no more than 25 stress cycles having a S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of Section III Appendices, Mandatory Appendix I.

When the Component Design Specification requires computations to demonstrate compliance with specified Service Limits, the Component Design Specification provides information from which Service Loadings can be identified (pressure, temperature, mechanical loads, cycles, or transients).

Design Pressure - The specified internal and external Design Pressure is not to be less than the maximum difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the most severe loadings for which the Level A Service Limits are applicable.

The Design Pressure includes allowances for pressure surges.

Design Temperature - Except as otherwise defined in ASME BPVC, Division 1, NB-3112 for Class 1 components, the specified Design Temperature is not less than the expected maximum mean metal temperature through the thickness of the part considered for which Level A Limits are specified.

Design Mechanical Loads - The specified Design Mechanical Loads are in accordance with NCA-2142.1C.

3.6.3.6.4 Test Loadings

Test Pressure - The specified internal and external test pressures are as required by the ASME BPVC, Section III, Division 1.

Test Loads - Loads due to other types of required tests are included as required by the ASME BPVC, Section III, Division 1.

Test Temperature - Test temperature is defined to ensure that thermal effects are considered in test loads.

3.6.3.7 Loading Phenomena

Section 3.6.3.7 describes the types of load phenomena, that is considered for components, as applicable.

3.6.3.7.1 Flow Induced Vibration

Flow of fluids past objects creates local pressure disturbances, which exert forces on the object. These forces can cause dynamic responses depending on the forcing function and dynamic characteristics of the object. Flow induced vibrations have been noted in nuclear power plant systems, which produce vortex shedding (e.g., heat exchangers), pump (reciprocating or centrifugal), and thermodynamic instability conditions. Design changes are reviewed for potential FIV mechanisms, evaluating all modes of system operation including both normal and abnormal conditions. Requirements for vibration monitoring are not within the scope of this document.

FIV loads may be associated with Service Level A for those structures (e.g., reactor internals) where the loads exist during normal operation. For FIV loads associated with transients that are not considered part of normal operation, the FIV loads are evaluated as part of the alternative service level.

Vortex Shedding

Vortex shedding occurs at certain fluid velocities when a fluid flows past objects. The dynamic response is controlled by proper spacing of the support plates for the tube bundle. The vibration cannot be eliminated but it can usually be controlled. It is important that these cases consider all potential modes of component operation. Vortex shedding hydrodynamic mass effects are considered. Other components susceptible to flow induced vibration are pressure, flow, and temperature sensors, which encroach upon the flow stream.

Pressure Fluctuations

Pressure fluctuations in a vapor or gas-state fluid (e.g., steam) occur due to flow past branch piping connections and branch connected components (e.g., safety valve “bell chamber” resonance), flow through short radius elbow fittings that induce flow separation effects, flow passing through valve chambers, flow past sharp-edged in-line pipe components (e.g., orifices, weld joint backing rings, valve seat rings), or two or more individual flows entering a common header or drum that generates an acoustic response. These various flow disturbances generate acoustic waves that can travel forward and backward in a piping system. If of sufficient strength and at a component’s susceptible frequency, these acoustic resonances can cause cyclic fatigue and result in component failure.

Pumps create pressure fluctuations in a fluid system. In most system designs, these fluctuations are insignificant. However, the possibility exists that these fluctuations, coupled with unintentional but improper system or component structural characteristics, can cause resonant vibrational response in the system or component. Component structural characteristics are designed to assure a resonance value sufficiently high to avoid excitation by evaluated system fluid fluctuations. Pressure attenuation devices are used as applicable to significantly reduce the effects of this phenomenon.

Thermodynamic Instability

Under certain system design features and operating modes, fluid dynamic forces can be generated, which create large pressure variations. These have been noted in certain feedwater systems where a relatively cold fluid layer is in contact with a relatively hot steam region; under certain operating modes significant water-hammer-type phenomena have occurred causing a breach of the pressure-retaining boundary.

3.6.3.7.2 Rapid Valve Closure or Opening

Extremely rapid valve closure or opening in a fluid system can create large pressure waves which can propagate through a piping system and into connected components. This rapid motion could be caused by operating characteristics of the valve (e.g., stiffness of diaphragm in pneumatic operators), the fluid flow forces acting on the valve parts during all modes of operations.

For example, TSV closure has been identified as being capable of generating large pressure waves which could cause significant dynamic response. Prior to TSV closure, saturated steam flows through main steam piping at nuclear boiler rated pressure and mass rate. Steam flow to the turbine comes to a stop at the instant the turbine stop valve closes. The flow of steam travels in the main steam line through the vessel nozzle and into the vessel. This results in a compressive acoustic load on steam dryer outer hood, as well as steam impingement load on steam dryer outer hood. Additionally, repeated reflections of the compression wave in the main steam line generate time-varying forces in the main steam piping. System, components, and structures in the Reactor Building, Steam Tunnel and Turbine Building may be affected.

3.6.3.7.3 Isolation Condenser Operation

The thermal effects associated with operation of ICS and the loads such as pressure resulting from operation of ICS are considered. Loads associated with the breaks of ICS high pressure lines in the pool are considered. The major loads imposed on ICS result from:

- Sudden reactor isolation at power operating conditions
- During station blackout (i.e., unavailability of all alternate current power)

- Failure to Scram
- LOCA

3.6.3.7.4 Failures of High-Energy Fluid System Piping

The effects of postulated pipe breaks in high-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

3.6.3.7.5 Failures of Moderate-Energy Fluid System Piping

The effects of postulated pipe cracks in moderate-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

3.6.3.7.6 Fuel Lift Loads

Fuel lift is the postulated process under which a combination of vertical motion of the RPV support, scram uplift forces on the fuel assemblies and vertical hydraulic forces result in fuel assemblies lifting off from their seating surfaces on the fuel support. The reaction load of the fuel on the core support structures is considered.

3.6.3.8 Safety Class Functional Criteria

For any normal or off-normal design condition event, safety class equipment and piping can accomplish the safety class functions as required by the event and incurring no permanent changes that could deteriorate the ability to accomplish safety class functions as required by any subsequent design-condition event.

For any emergency or faulted design-condition event, safety class equipment, and piping are capable of accomplishing their safety class functions as required by the event, but repairs could be required to ensure their ability to accomplish safety class functions as required by any subsequent design-condition event.

3.6.3.9 Reactor Pressure Vessel Assembly

The reactor vessel assembly includes: the RPV pressure boundary out to and including the nozzles, the RIV's, and the housings for FMCRD and nuclear instrumentations. The RPV assembly is an ASME BPVC Section III, Division 1, Class 1.

The feedwater nozzle design does not allow incoming feedwater flow to have direct contact with the nozzle bore region. A double thermal sleeve design provides protection against thermal cycling on the nozzle bore. The ICS Condensate Return nozzles use a similar single thermal sleeve design to mitigate thermal cycling of the nozzle bore during initial IC train operation when accumulated condensate is draining.

The stress analysis is performed on the RPV for various plant operating conditions (including faulted conditions) by using elastic methods, except as noted in Subsection 3.6.1.4.3. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are provided in Table 3.6-2.

The RPV internals are classified in Chapter 3, Section 3.2, and Appendix 3A. Complete stress reports on these components are prepared in accordance with the ASME BPVC Section III, Division 1, requirements.

3.6.3.10 Main Steam Piping

The MS piping trains extending from the outboard MSRIV to and including Seismic Interface Restraints (SIR) that are outboard of the MSCIVs are designed and constructed in accordance with the ASME BPVC Section III Division 1 rules for Class 2 Nuclear Components. Stresses are calculated on an elastic basis for each service level and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. Table 3.6-11 shows the specific load combinations and acceptance criteria for Class 2 piping that apply to this piping.

The MSCIVs, are designed and constructed in accordance with the ASME BPVC III Division 1, NCD-3500 requirements for Class 2 components.

The MS system piping extending from the outboard SIR to the turbine stop valve is constructed in accordance with the ASME B31.1 Criteria.

3.6.3.11 Other Components

3.6.3.11.1 Isolation Condenser System (ICS) Condenser and Piping

The ICS piping inside the primary containment between the RPV and the Isolation Condenser Heat Exchanger is designed and constructed in accordance with the ASME BPVC Section III Division 1 requirements for Class 1 piping. The isolation condenser and piping outside containment are designed and constructed in accordance with ASME BPVC Section III Division 1 Class 2 requirements.

3.6.3.11.2 CUW System Heat Exchangers

The CUW heat exchangers (regenerative) are not part of a safety system. However, the heat exchangers are Seismic Category NS equipment. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the CUW System heat exchanger components.

3.6.3.11.3 SDC System Pump and Heat Exchangers

The SDC heat exchangers (nonregenerative) are not part of a safety system. However, the pumps and heat exchangers are Seismic Category NS equipment respectively. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the SDC System pump and heat exchanger components.

3.6.3.11.4 ASME BPVC Section III Division 1, Class 2 and 3 Vessels

ASME BPVC Section III Division 1, Class 2 and 3 vessels are constructed in accordance with the ASME BPVC Section III Division 1. The analysis of these vessels is performed using elastic methods.

3.6.3.11.5 ASME BPVC Section III Division 1, Class 1, 2 and 3 Valves

ASME BPVC Section III Division 1, Class 1, 2, and 3 valves are constructed in accordance with the ASME BPVC Section III Division 1.

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The analysis of these valves is performed using elastic methods. Refer to Subsection 3.6.3.9 for additional information on valve operability.

3.6.3.11.6 ASME BPVC III Division 1, Class 1, 2 and 3 Piping

ASME BPVC Section III Division 1, Class 1, 2 and 3 piping is constructed in accordance with the ASME BPVC Section III Division 1. For ASME BPVC Section III Division 1, Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME BPVC Section III Division 1, and fatigue usage is determined. For ASME BPVC Section III Division 1, Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. If a NB-3600 analysis is performed for ASME BPVC Section III Division 1, Class 2 or 3 pipe, all analyses required for ASME BPVC Section III Division 1, Class 1 pipe as specified in this document and the ASME

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 values.

The major safety class active valves are the RIVs, Condensate Return Valves and CIVs. These valves are designed to meet the ASME BPVC Section III Division 1 BPVC requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is provided individually below.

3.6.3.13 Main Steam Containment Isolation Valves

The MSCIVs are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a Design Basis Accident (DBA) and DBE.

3.6.3.14 Other Active Valves

Other safety class active valves are ASME BPVC Section III Division 1 Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves operate during a dynamic seismic and other RBV event.

3.6.3.14.1 Procedures

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components, which are depended upon to cause the valve to accomplish its intended function, are developed to assure these functions are accomplished.

3.6.3.14.2 Tests

Prior to installation of the SC1 valves, the following tests are performed at the factory facility as required in the field:

- Shell hydrostatic test to the ASME BPVC Section III Division 1 requirements
- Seat leakage tests
- Obturator hydrostatic test
- Functional tests to verify that the valve opens and closes within the specified time limits when subject to the design differential pressure

The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.6.3.14.3 Check Valves

Due to the simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

- Stress analysis including the dynamic loads where applicable
- In-shop hydrostatic tests
- In-shop seat leakage test

3.6.3.15 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for simple devices (e.g., relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a simple device, that is an integral part of an assembly, may be subjected to the same dynamic load tests while in an operating condition. Thus, the performance of a simple device may be monitored during the test. However, for complex panels, such a test is not always practical. In this situation, the following alternate approach may be followed.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar but inoperative devices installed, is vibration tested to determine if the panel response accelerations. Installing the non-operating devices assures that the test panel has representative structural characteristics of a production panel. The accelerations are measured by accelerometers installed at the device attachment locations. The accelerations are less than the levels at which the devices were qualified. If the acceleration levels at all the device locations are found to be less than the levels to which the devices are qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices are requalified to the higher levels.

3.6.3.16 Design of Pressure Relief Devices

The NBS system does not utilize safety or relief valves for overpressure relief. During normal operation, the mainsteam flow to the turbine is throttled to control system pressure. Chapter 6, Section 6.2 describes the method of overpressure relief.

3.6.3.17 Component Supports

The establishment of the design/service loadings and limits is in accordance with the ASME Section III, Division 1, Article NCA-2000 and Subsection NF. These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events. The combination of loadings and stress limits are included in the Design Specification of each component and support.

ASME Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops, Pipe whip restraints are not considered as pipe supports.

The design of bolts for component supports is specified in the ASME BPVC III Division 1, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

3.6.3.18 Piping Supports

Supports and their attachments for ASME BPVC Section III Division 1 Class 1, 2, and 3 piping are designed in accordance with Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed as specified in Section 3.5.

The design of supports for the non-nuclear piping satisfies the requirements of ASME B31.1 Power Piping Code, Paragraphs 120 and.

3.6.3.19 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a SC1 linear type component support in accordance with the requirements of ASME BPVC Section III Division 1 Subsection NF. The stabilizer provides a reaction point near the upper end and lower end of the RPV to resist horizontal loads caused by effects such as earthquake, pipe rupture, and RBV. The design loading conditions, and stress criteria and the calculated stresses will meet the ASME BPVC Section III Division 1 allowable stresses in the critical support areas for various plant operating conditions.

3.6.3.20 Floor-Mounted Major Equipment

The condenser modules in the Isolation Condenser System (ICS) are analyzed to verify the adequacy of their support structure under various plant operating conditions. The analysis applies the maximum sheer, 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 DECs 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.

<u>Static Loads</u>

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 DECs 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 B31.1piping system.

The pipe supports attached to the ASME B31.1piping 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 DECs 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.

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.

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.

Table 3.6-1: Applicable Pressure Boundary Codes and Standards

Code or Standard Number	Title/Description
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

Table 3.6-2: Load Combinations and Acceptance Criteria

Plant Event / Event Combination	Service Loading Combination(1)(2)(3)(10)	Comments	ASME Service Level(4)
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(11) for Level B	B(6) (7)
Design Basis Accident	(a) N + DBA_A (b) N + DBA_B Loadings	OPG/CSA requirement for DBE(11) for Level C	C
Design Extension Condition	(a) N(5) + DEC_A (b) N(5) + DEC_B	OPG/CSA requirement for CLE(11) for Level D	D
Test(9)	$P_t + T_t + D_t$		Testing Limit(8)

(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

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 to 1.67) *DBE.

Table 3.6-3: Comparison of Event Frequency to Plant Conditions and Service Loadings

Design Condition (DC)	ASME Service Level	Quantitative Frequency (F) (1/year)
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.	< 1E-02
Design Basis Accidents (DBAs) (DC-3)	C; - incidents of low frequency – infrequent loadings causing no significant loss of integrity.	1E-02 > F ≥ 1E-05
Design Extension Conditions (DECs) (DC-4)	D; - incidents of extremely low frequency loadings associated with beyond design basis accidents.	F ≤ 1E-05

Table 3.6-4: Normal Operating Events (DC-1)

Description	Number of Cycles/60 Years
Boltup	72
Startup	200
Turbine Roll and Increase to Rated Power	200
Daily/Weekly Load Reduction and Recovery	20,805
Rod Sequence/Pattern Change	30
Rated Power Operation	-
Reduction to 0% Power	200
Hot Standby	200
Shutdown	200
Vessel Flooding/Shutdown Cooling	72
Unbolt	72
Refuel	72

Table 3.6-5: Test Events (DC-1)

Description	Number of Cycles/60 Years
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

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

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

Description	Number of Cycles/60 Years
Improper Startup – Hot Cleanup Water System	1 (freq $\leq$ 0.1)
Turbine Generator Trip. Load Rejection – Without Bypass	1 (freq $\leq$ 0.1)
Reactor Overpressure – Backup Scram	1 (freq$\leq$ 0.1)
Shutdown due to Inadvertent Isolation Condenser System (ICS) Initiation	1 (freq $\leq$ 0.1)
Inadvertent Sodium Pentaborate Injection	1 (freq $\leq$ 0.1)
Excessive Cooldown Rate	2 (freq $\leq$ 0.1)

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

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

Table 3.6-9: Summary of Cycles of Events

Event #	Description	Design Basis Number of Cycles
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

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 (2)(3)	Acceptance Criteria per ASME Code(1)(4)
Design	PD + WT	NB-3652
Service Level A and B(5)	PP, TE, ∆T1, ∆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.

Table 3.6-11: Load Combinations and Acceptance Criteria for ASME BPVC III Division 1 Class 2 and 3 Piping Systems

Service Level	Load Combination for all Terms(1)(2)(3)	Acceptance Criteria per ASME Code(4)(5)
Design	PD + WT	NCD-3652
A & B	TE	NCD-3653.2
A & B	Single Non-repeated Anchor Movement	NCD-3653.2
A & B	PD + WT + TE	NCD-3653.2
B	PP + WT + AOO Where AOO includes but is not limited to: TSV	NCD-3653.1
C	PP + WT + DBA Where DBA includes but is not limited to: LOCA DBE	NCD-3654.2
C	PP	NCD-3654.1
D	PP + WT + DEC Where DEC includes but is not limited to: SRSS (DBE + TSV) SRSS (DBE + LOCA)	NCD-3655
D	PP	NCD-3655

(1) TSV loads are used for MS lines only

(2) Where:

   a. WT = Dead Weight
   b. PD = Design Pressure
   c. PP = Peak Pressure or the Operating Pressure Associated with that transient
   d. DBEI = Design Basis Earthquake (inertia Effect)
   e. DBED = Design Basis Earthquake (Anchor Displacement Loads)
   f. DBE = Design Basis Earthquake includes both DBEI and DBED which are combined using SRSS method
   g. AOO = Anticipated Operational Occurrence
   h. DBA = Design Basis Accident
   i. DEC = Design Extension Condition

(3) LOCA is intended to represent loads and the appropriate combination of loads resulting from postulated line breaks including but not limited to Acoustic Inertial (ACI), JR, and JI loads

(4) ASME BPVC SECTION III NCD-2021

(5) Stress limits are reduced for piping locations exempt from pipe break consideration.

Table 3.6-12: Load Combinations and Acceptance Criteria for Non-Safety Class Power Piping Systems

Description	Load Combination	Acceptance Criteria per ASME Code(2)
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 DECs are generally considered outside of the scope of a qualification program, guidance is provided for demonstrating with reasonable assurance that equipment credited to perform under DEC conditions will survive to perform its function. See Subsection 3.9.3.5 for consideration of a Beyond-Design Basis Earthquake (BDBE) and Subsection 3.9.4.1 for Environmental Qualification considerations.

The focus of this section is on qualification of mechanical and electrical equipment. Mechanical equipment consists of items of a facility including pumps, valves, vessels, and piping whose function is required to ensure the safe operation or safe shutdown. Electrical equipment consists of all electrical power and Instrumentation and Control (I&C) equipment, which includes all analog (non-digital) and digital I&C components. Computer-based I&C equipment is a subset of digital I&C components.

Qualification of civil structures is covered in Section 3.3.

3.9.3 Seismic

3.9.3.1 General

Seismic qualification is a subset of equipment qualification that is the verification, through testing, analysis, or other methods, of the ability of an SSC to perform its intended function during and/or following a designated earthquake. The dynamic loads of Reactor Building Vibrations (RBVs) and events caused by hydrodynamic loads are also considered. Seismic and dynamic qualification of BWRX-300 equipment and associated supports meets the requirements and recommendations of the CSA N289 series (References 3.9-2 To 3.9-6) as endorsed by CNSC REGDOC-2.5.2 (References 3.9-1), and IEC/IEEE 60980-344 (Reference 3.9-7).

The requirement for seismic qualification is based on the seismic categorization of SSC and the earthquake level they are required to withstand during and/or after the seismic event. Seismic categorization of BWRX-300 SSC is described in Section 3.2. Seismic Category A and Seismic Category B SSC are most important and have the most stringent requirements for functional integrity during and following a seismic event. Per regulatory guidance of CNSC REGDOC-2.5.2, Section 5.13.1 (Reference 3.9-1), SSC that are classified as Seismic Category A and Seismic

Category B are seismically qualified to withstand the effects of a DBE. The site-specific DBE is defined in Subsection 3.3.1.

BWRX-300 equipment Seismic Categories are identified in Appendix 3A Table 3.12-1. Seismic Categorization of Structures is provided in Section 3.3, Table 3.3-1.

3.9.3.2 Methods for Seismic Qualification

Seismic and dynamic qualification of equipment and associated supports are accomplished by test, analysis, or a combination of testing and analysis. Seismic and dynamic qualification of equipment and associated supports designated as SC1 is accomplished by testing. Seismic and dynamic qualification of equipment and associated supports designated as SC2 may be accomplished by analysis or a combination of testing and analysis.

Qualification by actual seismic experience (also referred to as seismic qualification by similarity), as described in IEC/IEEE 60980-344 (Reference 3.9-7) and CSA N289.1 (Reference 3.9-2), is also utilized as appropriate considering the limitations identified in CSA N289.1, Annex D.3 (Reference 3.9-2).

The selection of qualification method to be used is largely a matter of engineering judgment for cases where testing is not required. When both test and analysis are defined as acceptable methods, the deciding factors considered (as applicable) for choosing between tests or analysis include magnitude and frequency of seismic and RBV dynamic loadings, environmental conditions associated with the dynamic loadings, nature of the safety category function(s), size and complexity of the equipment, dynamic characteristics of expected failure modes (structural or functional), and partial test data upon which to base the analysis.

Tests or analyses of assemblies are preferable to tests or analyses on separate components (e.g., a motor and a pump, including the coupling and other appurtenances, should be tested or analyzed as an assembly), unless deemed not practical. Equipment that has been previously qualified by means of tests and analyses equivalent to those required for the current qualification program are used if proper documentation of such tests and analyses is available.

For equipment defined as requiring test for qualification, analysis by similarity may be used if similar equipment is being or has been qualified by test.

3.9.3.2.1 Testing

Testing of BWRX-300 SSC for seismic qualification is conducted in accordance with CSA N289.4 (Reference 3.9-5) IEC/IEEE 60980-344 (Reference 3.9-7).

Seismic qualification by testing is typically used for SSC that will be performing an active function and are required to change state during or following a seismic event to perform a safety category function, while maintaining structural and/or pressure boundary integrity. Seismic testing can identify contact chatter or unauthorized change of state of contact in electrical and I&C components during seismic excitation.

The dynamic test sequence includes as applicable, vibration conditioning, exploratory resonance search, low-level earthquake loading (one-half DBE) including Reactor Building Vibrations (RBV) dynamic loads and the DBE loading including RBV dynamic loads.

Dynamic tests are performed with the equipment subjected to nominal operating service conditions. Significant, normal operating loads such as electrical, mechanical, pressure, and thermal are included. Where normal operating loads cannot be included in the dynamic tests, supplemental analysis is used to qualify the equipment for those effects. If there is any dynamic coupling due to interacting equipment, it is considered.

For equipment located in multiple locations, the enveloping upper bound seismic condition limits are used to eliminate the need for multiple qualification tests, unless otherwise specified.

Resonance Tests

When required, exploratory resonance search tests (such as sine sweeps or random vibration) are used for equipment to help determine the method of test or analysis that would be best for qualification and/or determine the dynamic characteristics such as the resonance frequencies of the equipment, mode shapes and damping values.

Sine sweep resonance search is the preferred method and is performed by running a continuous sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude at which resonance can be determined.

Resonance searches may be performed prior to and after the seismic test to determine any shifts in frequency caused by testing.

If resonance frequencies are present, the transmissibility between the input and the location of the equipment is determined by measuring the accelerations at the equipment location and calculating the magnification between it and the input.

Floor-mounted frequency testing can be used as another method to determine the resonance or natural frequencies for equipment.

Seismic Input Motion

Dynamic load conditions are simulated by testing, using independent, random multi-frequency input or single frequency input motion (within equipment capability) over the frequency range of interest.

Acceptable justification for use of single frequency input includes, but is not limited to:

1. The characteristics of the required input motion are dominated by one frequency.
2. The anticipated response of the equipment is adequately represented by one mode.
3. The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelop the corresponding response spectra of the individual modes.
4. The time phasing of the inputs in the vertical or horizontal directions will be such that a purely rectilinear resultant input is avoided.

The actual input motion used during testing, for both multi and single frequency, envelops the applicable input motion (floor, wall, response, etc.) at the location(s) of the equipment under test.

When the equipment is qualified by dynamic test, the In-Structure Response Spectra (ISRS) or time histories, developed from the results of Soil-Structure Interaction (SSI) analyses as described in Section 3.3.1.2.7, representing the in-structure seismic response of the attachment point is used in determining required response spectra of input motion used for the test.

For the case of equipment having multiple supports with different dynamic motions, the effects of the multiple support attachment points must be considered in the dynamic qualification and can be accounted for by selecting an upper bound envelope of all the individual response spectra for these locations to calculate the maximum internal responses applicable to the equipment, unless otherwise specified.

Past testing demonstrates that Seismic Category A electrical equipment has critical damping ratios equal to or less than 5%. Hence, the required response spectra at 5% or less critical damping ratio are developed as input to the equipment base, unless identified otherwise.

Seismic Test

The preferred test method for seismic qualification is shake table testing. Seismic testing is performed in a manner that demonstrates dynamic response characteristics and acceptability of the test specimen to withstand and maintain its function as required during the expected level of shaking. Test requirements are normally specified in the form of required response spectra at a specified damping value and confirmed by a Test Response Spectra (TRS) generated from the table motion.

The seismic test for DBE produces a TRS that envelops the applicable portion of the required response spectra as defined in the test specification (typically by a factor of 1.1) per CSA N289.4 (Reference 3.9-5), The approach is to apply 10% to the acceleration of the ISRS, developed from the results of SSI analyses as described in Section 3.3.1.2.6, which meets the recommendations of IEC/IEEE 60780-323 (Reference 3.9-8).

Testing for low-level earthquake loading and RBV dynamic loads is performed to demonstrate that the low-level earthquake loads combined with RBV dynamic loads do not degrade the continued structural and functional integrity of the equipment.

Testing for DBE loading and RBV dynamic loads are performed to demonstrate that equipment would perform its intended function(s) through DBE combined with RBV dynamic loads.

For both low-level earthquake and DBE seismic test runs, the input excitation TRS is required to envelop the specified required response spectra levels in accordance with CSA N289.4 (Reference 3.9-5) and Section 9 of IEC/IEEE 60980-344, (Reference 3.9-7).

If the TRS do not meet the requirements (i.e., do not envelop the required response spectra, do not demonstrate stationarity, do not demonstrate statistical independence) for the seismic test run, the test run is documented as unacceptable, adjustments may be required, and then the test is repeated.

Alternatively, per Clause 5.1.2.2.4 of CSA N289.4 (Reference 3.9-5), for acceptance in cases where TRS does not envelop required response spectra, the following criteria are applied:

- The number of points below the required response spectra shall not exceed 5
- The points shall not fall below the required response spectra by more than 10%
- Any two points below the required response spectra shall be at least 1 octave apart
- The points adjacent to the points that fall below the required response spectra shall be at least 10% above the required response spectra

For equipment that is subjected to vibration in its in-service condition, vibrational aging to its end of life condition must be completed prior to seismic testing (both low-level earthquake and DBE load tests).

For seismic qualification, the seismic input consists of five one-half DBE amplitude events (low-level earthquakes) followed by one DBE event. Alternatively, in accordance with Annex E of IEC/IEEE 60980-344 (Reference 3.9-7), a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half DBE events may be used followed by one full DBE event; however, in this case the amplitude shall not be below the minimum of one-half the DBE input motion.

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).

The decision on using dynamic versus static analysis is typically defined based on whether the equipment is rigid or flexible.

If the fundamental frequency of the equipment is above the input excitation frequency (cutoff frequency of required response spectra) the equipment is considered rigid.

The search for the natural frequency is done analytically, if the equipment shape can be defined mathematically, or by prototype testing.

If the equipment is determined to be a rigid body (i.e., shown to have no resonance frequency within the expected frequency range) the static analysis method is able to be applied in place of dynamic analysis.

If the equipment is determined to be flexible (i.e., with the fundamental frequency of the equipment within frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied, unless justified otherwise.

If it is determined that either dynamic or static analysis can be used, in general, the choice of the analysis is based on the expected design margin, as the static coefficient method is more conservative than the dynamic analysis method.

For static analysis, the dynamic forces on each component can be obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment.

A static coefficient analysis may also be used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the required response spectra at the equipment mounting location, at a conservative and justifiable value of damping.

Both types of analyses verify integrity of the equipment is maintained under low-level earthquake loads including appropriate RBV dynamic loads in combination with normal operating loads and normal operating and DBE loads including appropriate RBV dynamic loads, unless otherwise justified.

See Section 3.3.1.3 for additional details and discussion of Seismic Analysis of Seismic Category A and B Subsystems.

3.9.3.4 Seismic Qualification by Combined Testing and Analysis

Qualification by combined testing and analysis is used as a method for qualification for complex or large equipment where it is not practical to test the entire assembly or it is too large to be tested at once, unless another method of qualification is justified.

One method of combined qualification is to use a representative prototype portion or scaled-down prototype of the assembly that is subjected to type testing. The data from the type testing is then used to develop and validate an analytical model of the prototype. The prototype analytical model is then extrapolated to represent the larger assembly and then using the results to justify qualification of the equipment based on prototype testing.

A second method of combined qualification is to mount the full assembly to a rigid floor to simulate service mounting and then a portable shaker test (or an impact or pull test if justified) is performed to excite the natural or resonance frequencies of the specimen. The amplification of resonance motion is used to determine the appropriate modal frequency and damping for a dynamic analysis of the equipment.

For equipment with multiple site configurations the combined qualification method can be applied to reduce the number of configurations to be tested. In this case, an evaluation must be performed to determine the enveloping “worst-case” configuration(s), which is then tested. Analysis is then used to justify the various configurations based on the “worst-case” configuration(s).

The combination method can be used for qualification of larger electrical equipment support assemblies containing Seismic Category A or B equipment. For this case, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range. If the support is determined to be free of natural frequencies in the critical frequency range, then it is assumed to be rigid and a static analysis is performed and calculations of transmissibility and responses to varying input accelerations are determined to see if Seismic Category A or B equipment mounted in the assembly would operate without malfunctioning.

3.9.3.5 Beyond Design Basis Earthquake

REGDOC-2.5.2 Section 7.13 (Reference 3.9-1) states that for a beyond-design-basis earthquake (BDBE), demonstration that there is a high confidence of low probability of failure (HCLPF) of the SSC that are credited to function during and after the event. This demonstration need not be seismic qualification by testing. BDBE is identified as a Checking Level Earthquake (CLE). Typically, the CLE (as discussed in Section 3.5.6.1.2) is considered a DEC. DECs 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 DECs 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 DECs 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, conservatisms 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 he 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.

Consistent with CNSC REGDOC-2.5.2 (Reference 3.9-1), EMI/RFI is addressed through recognized industry standards. NRC Reg Guide 1.180 (Reference 3.9.-12) provides appropriate guidance for the EMC testing, describing methods and procedures considered acceptable for demonstrating EMC compliance based on the endorsement of IEC standards IEC 61000-2 / 4 (References 3.9-13 and 3.9-14), Military Standards MIL-STD (Reference 3.9-15) EPRI Topical Report TR-102323 (Reference 3.9-16) and IEEE Standard 627 (Reference 3.9-17) for test methods consistent with specific equipment requirements.

Chapter 2, Section 2.2.9 characterizes the site-specific electromagnetic hazards for which the design must consider and for which EMC qualification must address.

Chapters 7 and 8 describe the design of the I&C systems and the Electrical systems, respectively. As part of the design process, layout strategies are developed to ensure that the design considers interaction between SSC, and as the design is constructed, elements such as grounding and shielding are incorporated to meet the EMC/EMI standards (prior to testing).

Chapters 7, Subsections 7.3.1.3.1, 7.3.2.3.1, and 7.3.4.3.1 discuss design and quality measures for I&C systems as they relate to qualification measures that confirm I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the plant state and for the area in which they are located. Chapter 7, Table 7.1-1 provides System and Equipment standards to be followed in the design that ensures qualification measures are applied.

Chapter 8, Section 8.6 provides electrical system design information on grounding and EMC. Chapter 8, Section 8.1.1.2 describes how electrical systems are designed to accommodate grid disturbances. The electrical design includes considerations for the environmental conditions postulated for plant states in the areas in which components are located and credited to function.

The standards referenced provide detailed test conditions to ensure equipment is tested in the environments in which they are expected to function and provide post-installation practices for maintaining qualification including handling and storage requirements.

3.9.6 Specific Equipment Requirements

Specific equipment categories may have additional requirements not applicable generically across all qualification programs. The Electrical and I&C equipment must meet the guidance provided in CSA N289 series (References 3.9-2 through 3.9-6) and the CSA N290 series standards (Reference 3.9.18 through Reference 3.9-22).

3.9.6.1 Mechanical Equipment

Safety Class mechanical equipment, which has the sole safety category function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME Boiler and Pressure Vessel Code, Section III (Reference 3.9-23), is considered qualified as specified in CSA N290.13 (Reference 3.9-9).

Mechanical equipment can be qualified by presenting historical performance data if it is demonstrated that the equipment satisfactorily sustains dynamic loads which are equal to or greater than those specified for the equipment and that the equipment performs a function equal to or better than that for which it is specified.

For mechanical equipment where the loading under normal service is more severe than loading under DBA, then the loading under normal service must be considered in addition to the loading under DBA by test and/or analysis.

For mechanical equipment, the loading and capability under DBA conditions is analyzed in the qualification process to establish the suitability of materials, parts, and equipment needed for safety category functions, and to verify that the design of such materials, parts, and equipment is adequate.

The qualification of mechanical equipment includes, as applicable, materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment, the environmental conditions and process parameters for which this equipment must be qualified, non-metallic material capabilities, and the evaluation of environmental effects.

In addition, the qualification guidance provided in ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, (Reference 3.9-24), is considered for qualification of active mechanical equipment. Mechanical pipe supports of SC1 equipment that are susceptible to environmental degradation are seismically and environmentally qualified.

3.9.6.2 Electrical Equipment

Additional qualification guidance is considered for specific electrical equipment, if applicable, as follows:

- SC1 Batteries and their supporting element – IEEE 535 (Reference 3.9-25)
- SC1 Transformers IEEE 638 – (Reference 3.9-26)
- Static battery chargers and inverters – IEEE 650 (Reference 3.9-27)
- Electric penetration assemblies – IEEE 317 (Reference 3.9-28)
- SC1 Actuators – IEEE 382 (Reference 3.9-29)
- SC1 Continuous duty motors – IEEE 334 (Reference 3.9-30), as endorsed by Reg Guide 1.40 (Reference 3.9-31)
- SC1 Motor Control Centers (MCCs) – IEEE 649 (Reference 3.9-32)
- For the electrical equipment described above, excluding motors, the EMC qualification guidance provided in Reg Guide 1.180, (Reference 3.9-17) is considered

3.9.6.3 Instrumentation & Control Equipment

Additional qualification guidance is considered for specific I&C equipment, if applicable. For example, control boards, panels, and racks classified as SC1 components utilize IEEE 420, (Reference 3.9–33) for their qualification program.

Qualification of computer-based I&C systems is in accordance with CNSC REGDOC-2.5.2, (Reference 3.9-1), CSA N290.13 (Reference 3.9-9), and IEEE 7-4.3.2 (Reference 3.9-34) which is consistent with the EMC requirements specified in Reg Guide 1.180 (Reference 3.9-12) and described in Subsection 3.9.5.

When computer based I&C systems environmental type testing is performed:

1. The system under test demonstrates that it functions and performs with safety software that has been validated and verified and is representative of the software to be installed in-service.
2. The testing demonstrates performance of all safety category functions that may be impacted by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, that are tested during verification and validation testing, are not required to be tested unless their outputs exercise different hardware components which may be impacted by environmental conditions.

3. The testing exercises all portions of the system that are necessary to accomplish the safety category functions and those portions whose operation or failure could impair the safety category functions.
4. The testing confirms the response of digital interfaces and verify that the design accommodates the potential impact of environmental effects on the overall response of the system.

The testing of a complete system is preferred. When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety performance.

3.9.6.4 Cables, Raceways, Supports, etc.

For qualification of SC1 cables, the qualification guidance provided in CSA N290.13, (Reference 3.9-10) and IEEE 383 (Reference 3.9-35) are considered.

Supports (hangers) that support trays or conduit that carry safety circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628 (Reference 3.9-36).

Supports used for Non-Safety Class raceway (conduit and cable tray) in Seismic Category A structures are analyzed to withstand the effects of a DBE and evaluated for seismic interaction as applicable.

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, (Reference 3.9-37) as endorsed by Reg Guide 1.156, (Reference 3.9-38) for their qualification program.

3.9.6.5 Line-Mounted Equipment

Guidance in IEEE 572 (Reference 3.9-37) and IEC/IEEE 60980-344 (Reference 3.9-9.) identifies that special consideration is required for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition will occur as a result of the piping or duct system.

Guidance and further clarification for special considerations for line-mounted equipment is provided in IEEE 572 (Reference 3.9-33) and IEC/IEEE 60980-344 (Reference 3.9-8) as well as IEEE 382 (Reference 3.9.10.11-29).

3.9.7 References

3.9-1 CNSC Regulatory Document REGDOC-2.5.2, “Design of Reactor Facilities: Nuclear Power Plants.”

3.9-2 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants,” CSA Group.

3.9-3 CSA N289.2, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants," CSA Group.

3.9-4 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.

3.9-5 CSA N289.4, “Testing procedures for seismic qualification of nuclear power plant structures, systems, and components.” CSA Group.

3.9-6 CSA N289.5, “Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities,” CSA Group.

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.

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."

Table 3.9-1: Harsh Environment Parameter Conditions

Parameter		Condition
Temperature		10°C above normal ambient and ≥ 50°C [1]
Pressure		>4 kPa(g) (or 10%) increase or decrease from normal ambient pressure due to a DBA [2]
Humidity		100% Relative Humidity or condensing steam conditions [3]
Submergence		Any [4]
Radiation	Non-electronic equipment	DBA Total integrated accident dose (TIAD) > 170 Gy (17 krad) [5]
	Electronic equipment	TIAD > 10 Gy (1krad) [6]
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.

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 does limits for Adults."

3.12-12 CSA N286.7, “Quality Assurance of Analytical, Scientific, and Design Computer Programs,” CSA Group.

3.12-13 CSA N288.2, "Guidelines for Calculating the Radiological Consequences to the Public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents," CSA Group.

3.12-14 CSA N288.1:14, “Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities,” CSA Group.

3.12-15 USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning."

3.12-16 USNRC IN96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly."

3.12-17 ANSI/ANS-5.1, “American National Standard Decay Heat Power in Light Water Reactors,” American Nuclear Society.

3.12-18 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.

3.12-19 ASME BPVC-III APP, “Section III - Rules for Construction of Nuclear Facility Components - Appendices,” American Society of Mechanical Engineers.

3.12-20 ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13," American Society of Mechanical Engineers.

Table 3.12-1: Preliminary BWRX-300 Classification List

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
NUCLEAR STEAM SUPPLY SYSTEMS				
Nuclear Boiler System				
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: • Shroud • Chimney • Core Support Ring and Legs • (Shroud Support) • Core Plate (and Core Plate Hardware) • Top Guide (and Top Guide Hardware) • Orifice Fuel Supports and Peripheral Fuel Supports • Control Rod Guide Tubes (CRGTS) • Non-Pressure Boundary Portion of Control Rod Drive Housings (CRDHs)	SC1	SCCV	B	A
Internal Structures: • Nuclear Instrumentation In-Core Guide Tubes • Non-Pressure Boundary Portion of In-Core Housings	SC1	SCCV	B	A
Internal Structures: • Chimney Head and Steam Separator Assembly • Steam Dryer Assembly • Feedwater Spargers • Head Vent Internal Piping • CUW Suction Piping • Nuclear Instrumentation In-Core Guide Tube Stabilizers • ICS Return Internal Piping	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

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

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
INSTRUMENTATION AND CONTROL SYSTEM				
SC1 Instrumentation and Control System	SC1	RB and CB	NA	B
SC2 and 3 Instrumentation and Control System				
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
RADIATION MONITORING SYSTEMS				
Process Radiation and Environmental Monitoring System				
Process Radiation and Environmental Monitoring System, Process Radiation Monitoring Subsystem				
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
Process Radiation and Environmental Monitoring System, Area Radiation Monitoring Subsystem				
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
Process Radiation and Environmental Monitoring System, Containment Monitoring Subsystem				
CIVs and inboard process piping	SC1	RB	B	B

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
Process Radiation and Environmental Monitoring System, Process Sampling Subsystem				
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
CORE COOLING SYSTEMS				
Isolation Condenser System				
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

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

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

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
REACTOR SERVICING EQUIPMENT				
Refueling Equipment and Servicing				
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
REACTIVITY CONTROL				
Boron Injection System				
Injection Pump	SC3	RB	D	NS

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
Control Rod Drive System/High Pressure Injection				
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 HCUs), pump discharge, drive header, and other piping not part of HCU)	SC3	RB	D	NS
Charging Water Piping and Valves (directly above HCUs)	SC3	RB	D	NS

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
DECAY HEAT REMOVAL				
ICS Pool Cooling and Cleanup System (ICC)				
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
Shutdown Cooling System				
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
Reactor Water Cleanup System				
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

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
Fuel Pool Cooling and Cleanup System (FPC)				
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
NUCLEAR FUEL				
Nuclear Fuel Supply	SC1	SCCV, RB	NA	A
RADIOACTIVE WASTE MANAGEMENT SYSTEMS				
Liquid Waste Management System (LWM)				
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
Solid Waste Management System (SWM)				
SWM Equipment	SC3	RWB	D	NS
Spent Resin Tank	SC3	RWB	D	NS
Sludge Tank	SC3	RWB	D	NS
Offgas System (OGS)				
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

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
POWER CYCLE SYSTEMS				
Condensate and Feedwater Heating System				
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
Condensate Filters and Demineralizers System				
Filters, demineralizers, bypass lines, valves, and related components	SC3	All	D	NS
All other system equipment	SCN	All	D	NS
Main Turbine Equipment				
Main Turbine Equipment and Subsystem	SC3	TB	D	NS
Non-Return Valves	SC3	TB	D	NS
Moisture Separator Reheater System				
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
Turbine Bypass System				
Components supporting Turbine bypass	SC3	TB	D	NS

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
Generator and Exciter				
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
Main Condenser and Auxiliaries				
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
Circulating Water System				
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
STATION AUXILIARY SYSTEMS				
Chilled Water Equipment				
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

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
Plant Cooling Water System				
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
Plant Pneumatics System				
Containment Penetrations & Isolation Valves	SC1	SCCV, RB	B	A/B
All other system equipment	SC3	ALL	D	NS
Hydrogen Water Chemistry				
All system equipment	SCN	TB	D	NS
Zinc Injection Passivation				
All system equipment	SCN	TB	D	NS
STATION ELECTRICAL SYSTEMS				
SC1 Electrical Distribution System				
All System Equipment	SC1	RB	NA	A
Standby Electrical Distribution System				
SC2 Components	SC2	RB, CB	NA	NS
SC3 Components	SC3	ALL	NA	NS
Normal Electrical Distribution System				
SC3 Components	SC3	ALL	NA	NS
All System Equipment	SCN	ALL	NA	NS
POWER TRANSMISSION SYSTEM				
Switchyard				
All system equipment	SCN	Switchyard	NA	NS
CONTAINMENT AND ENVIRONMENT CONTROL SYSTEMS				
Primary Containment				

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
Containment Inerting System				
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
Containment Cooling System				
Drain valves	SCN	SCCV	D	NS
All other system equipment	SC3	SCCV	D	NS
STRUCTURE AND SERVICING SYSTEMS				
Cranes, Hoists, and Elevators				
All system equipment	SCN	ALL	NA	NS
Heating Ventilation and Cooling System				
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
Fire Protection System (FPS)				
System components that support DL2 or DL4b functions (Piping, valves and sprinklers)	SC3	ALL	D	TBD

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
Equipment and Floor Drain System				
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
Water, Gas, and Chemical Pads				
Components required to provide standby diesel fuel oil storage and transfer	SC3	ALL	D	NS
All other system equipment	SCN	ALL	D	NS

NOTES:

1. SC determination and methodology is discussed in Subsections 3.2.1 and 3.2.2.
2. Location Codes:
   a. SCCV: Containment Vessel
   b. RB: Reactor Building
   c. TB: Turbine Building
   d. CB: Control Building
   e. RWB: Radwaste Building
   f. OO: Outdoors On-site
   g. OL: Any Other Location
   h. ALL: All locations
3. Quality group classifications is discussed in Subsection 3.2.4.
4. Seismic categories are discussed in Subsection 3.2.3. Any items classified as NS are subject to evaluations for Seismic Interaction as discussed in Subsection 3.2.3.1.
5. Structures, systems and components required to be designed in accordance with Radioactive Waste Management requirements from RG 1.143 for Category RW-IIa, shall meet the guidance of NRC Regulatory Guide 1.143, as applied to radioactive waste management systems, with regard to quality, seismic, and quality group requirements.
6. Other abbreviations.
   a. TBD: To Be Determined – classification information is to be provided later in the BWRX-300 design process
   b. NA: Not Applicable
7. Components classified as SC1 may be assigned to a Seismic Class lower than A or B provided they are of a fail-safe design such that the failure of those component(s) does not adversely affect the ability to achieve the safety function.
8. Although these components are not SC1, they are seismically qualified because they are credited with monitoring leakage of reactor coolant under the scope of Regulatory Guide 1.45 or are related to handling and storage of used nuclear fuel.

APPENDIX 3B – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES

3.13 Introduction

This appendix describes the major computer programs used in the analysis and design of the BWRX-300 Seismic Category structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3.12-18) and CSA N286.7-16, “Quality Assurance of Analytical, Scientific, and Design Computer Programs” (Reference 3.12-12).

GEH maintains an ISO 9001:2015, "Quality Management Systems - Requirements," International Organization for Standardization“ (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12, “Management System Requirements for Nuclear Facilities” (Reference 3.12-6).

3.13.1 ACS SASSI v4

Description: ACS SASSI is a finite element computer code on the Microsoft Windows PC platforms for performing 3D dynamic soil-structure interaction (SSI) analysis to analyze the effect of seismic ground motions on structures. The analysis is performed in the frequency domain using linear or equivalent-linear material properties for the structure and soil.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: ACS SASSI is used to perform seismic and static SSI and structure-soil-structure-interaction (SSSI) analyses, as applicable.

3.13.2 ANSYS v17

Description: ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

3.13.3 Ansys LS-Dyna v2021

Description: Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

Validation: This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

Extent of Application: Ansys LS-DYNA is used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

3.13.4 SSI-StressCoord v1

Description: The STRESS_POST program is an auxiliary program to post-process the ACS SASSI NQA V4 STRESS result binary database. The STRESS_POST program includes an ensemble of STRESS database processing functionalities which were customized for the GEH engineers for application to the BWRX-300 SMR seismic SSI analysis projects. The STRESS_POST customized program is based on specific implementations incorporated in the ACS SASSI NQA V4 User Interface (UI) capabilities, such as the CTVEC and the CTCCV commands, and existing STRESS binary database verification tools used in-house during the development over years of the STRESS module.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This STRESS_POST Program is used for post-processing the ACS SASSI STRESS binary databases for Integrated RB Walls and Floors in batch mode.

3.13.5 GT STRUDL

3.13.5.1 Description

GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

3.13.5.2 Validation

This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

3.13.5.3 Extent of Application

GT STRUDL is used to for the structural analysis and design of non-Seismic Category A structures.

APPENDIX 3C – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF MECHANICAL STRUCTURES, SYSTEMS AND COMPONENTS

3.14 Introduction

As discussed in Subsection 3.6.1.1, this appendix describes the major computer programs used in the analysis of mechanical SSC.. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.14.1 ANSYS v17

Description: ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

3.14.2 PBLE v1

Description: Steam Dryer Analysis

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: PBLE calculates the acoustic loads on a steam dryer based on measurements of pressure along the main steam lines or pressures measured directly on the face of the steam dryer. The loads are then used in a finite element model to calculate the stresses in the dryer.

3.14.3 SIMCENTER 3D Acoustics v2022

Description: Used for modeling dryer acoustic loads and instrumentation diagnostics. Simcenter 3D is a unified, scalable, open and extensible environment for 3D CAE with connections to design, 1D simulation, test, and data management. Fast and accurate solvers power structural, acoustics, flow, thermal, motion, and composites analyses, as well as optimization and multi-physics simulation.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent Of Application: SIMCENTER Finite elements acoustic software will be used to model and calculate acoustic wave propagation in fluid (steam, water) mediums.

3.14.4 GT STRUDL

Description: GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GT STRUDL will be used to perform structural analysis and qualification of supports.

3.14.5 HyperMesh

Description: HyperMesh is the market-leading, multi-disciplinary finite element pre-processor which manages the generation of the largest, most complex models, starting with the import of a CAD geometry to exporting a ready-to-run solver file. With its advanced geometry and meshing capabilities, HyperMesh provides an environment for rapid model generation.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HyperMesh is a tool which will be used to generate mechanical models for complicated mechanical components. This tool will serve as a pre-processor to build mesh models, no calculations get performed with Hypermesh.

3.14.6 ERSIN v3

Description: Piping Analysis Software. Secondary Response Spectra for control panels, equipment racks, etc.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: ERSIN is used to generate secondary response spectra for pipe and floor mounted equipment. Example applications include control panels, equipment racks, Main Steam Isolation Valves (MSIVs), Safety Relief Valves (SRVs), Hydraulic Control Units (HCUs), et cetera. ERSIN03P software has three input options: 1) card decks, 2) SAP software decks, and 3) PISYS software decks. ERSIN03P can be used with SAP version 4G07P (Ref. 5-1) and PISYS version 08P (Ref. 5-2) structure/piping models only. If a card input is used, a mass normalized mode shape is required. ERSIN03P is not applicable for axisymmetric analyses using a Fourier Decomposition technique.

3.14.7 RINEX Computer Program

Description: RINEX is a computer code used to interpolate and extrapolate amplified response spectra used in the response spectrum method of dynamic analysis. RINEX is also used to generate response spectra with non-constant model damping. The non-constant model damping analysis option can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13" (Reference 3.12-20) damping values.

Validation: Hand calculations and test cases analyzed are used to demonstrate the program's applicability and validity.

Extent of Application: This program is used to generate multiple damping spectra for piping.

3.14.8 PDA (Civil)

Description: Pipe Dynamic Analysis (PDA) Pipe Whip Restraint Analysis

Validation: This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

Extent of Application: GEH in-house program for calculating pipe whip response under postulated break conditions. Determines response for a standard configuration which utilizes U-type pipe whip restraint.

3.14.9 PIPESTRESS

Description: PIPESTRESS (developed under a Quality Assurance Program compliant with the ASME NQA-1 (Reference 3.12-18) standard along with 10 CFR 21, "Reporting of Defects and Noncompliance" (Reference 3.12-9) and 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" (Reference 3.12-10)) is a pipe stress and flexibility analysis program, used for the evaluation of structural response and stress levels of piping systems against the requirements of industry codes and standards.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The plant layout, isometric drawings, P&ID, PFD, etc. will be used to build the piping model in PIPESTRESS, then PIPESTRESS will calculate the displacement, force/moment and stress. This software has the piping information, pipe routing & system information for BWRX-300 & some equipment information.

3.14.10 FLOMASTER v2021.1

Description: Uses simulation to offer reliable & accurate solvers and solutions for fluids engineering

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: Simcenter Flomaster is a unique thermo-fluid system simulation software tool used to simulate thermo-fluid systems; facilitating upfront engineering to reduce cost and lead times in product development and maintenance. It has an extensive library of component models, pre-populated with reliable performance data, Flomaster allows fluid system design to start before CAD data is available and component suppliers have been selected.

3.14.11 Ansys LS-Dyna v2021

Description: Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: Ansys LS-DYNA will be used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

3.14.12 3KeyMaster v2021 (ICE/Plant Integration Engineering/Systems Engineering)

Description: Plant-wide physics-based simulation supporting engineering design options, confirmation, and future reactor operator training full scope simulator (FSS) in accordance with ANS Std 3.5.

Validation: This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

Extent of Application: 3KeyMaster is used to generate plant layout schematics & run test simulations for new plant setups through variable/parameter manipulation for OPG.

APPENDIX 3D – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF ELECTRICAL STRUCTURES, SYSTEMS AND COMPONENTS

3.15 Introduction

This appendix describes the major computer programs used in the analysis of electrical SSC. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.15.1 ETAP v2021.1 (ICE Systems/I&C Tech)

Description: Electrical Transient Analyzer Program (ETAP) is an electrical network modeling and simulation software tool used by power systems engineers to create an "electrical digital twin" and analyze electrical power system dynamics, transients and protection.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: ETAP is the Global Market and Technology Leader of power systems solutions for a broad spectrum of sectors including Generation, Transmission, Distribution, Transportation, Industrial, and Commercial. The most comprehensive and integrated model-driven solutions for design, simulation, analysis, optimization, monitoring, operation, and automation of electrical power systems.

3.15.2 LDRA (I&C Tech/ICE Systems)

Description: Liverpool Data Research Associates is a provider of software analysis, and test and requirements traceability tools for the Public and Private sectors and a pioneer in static and dynamic software analysis.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: LDRA is a tool used to perform unit/module testing on software functions and components. It allows us to create and store test cases so we can perform regression testing, and it also allows us to execute the test cases on the target hardware (in this case an ARM Cortex-A9 processor).

3.15.3 Quartus II (I&C Tech/ICE Systems)

Description: Tools that provide FPGA compiler, simulation, and programming capabilities.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: Quartus is a tool used to develop applications for programmable logic devices such as PLDs and FPGAs. Applications in this case means the logic that the device implements. For example, it could be logic that provides a 2 out of 3 votes, it could be something that processes digital communications such as our fibre optic links, etc. Included in the software is something called timing analysis, which is a methodology for ensuring the logic inside the device meets timing characteristics. It also includes support for a simulator. The simulator allows engineers to evaluate the functionality of their logic by specifying input and examining how the logic reacts (e.g., verify the correctness of the design). The simulator does not require a physical device.

APPENDIX 3E – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSES STRUCTURES, SYSTEMS AND COMPONENTS – NUCLEAR FUELS

3.16 Introduction

This appendix describes the major computer programs used in the analysis of nuclear fuels. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems**.** The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.16.1 EPRI: Acube v11

Description: Advanced cutset upper bound estimator

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use ACUBE to post-process result cutsets using a Binary Decision Diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cutsets and returns an estimate of the probability for a given top event using the BDD method. The BDD method is more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

3.16.2 EPRI: CAFTA v11

Description: CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The CAFTA software will be qualified to complete all designed functions within the software. The use of the CAFTA software will be acceptable for use as is. Note that the testing will not cover every possible variation or combination of use for the software but it will validate the software operates as intended for within the standard operating configuration of the software.

3.16.3 EPRI: MAAP v5

Description: The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licenced computer software - is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Assessment (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use MAAP to analyze reactor thermal-hydraulic and containment response to transients as well as severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

3.16.4 EPRI: PRAQuant v11

Description: Accident Sequence Quantification. In performing a fault tree based analysis it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use PRAQuant in the processing of the combined hazard model to generate a combined hazard cutset output. PRAQuant is a processing software to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation, etc.) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification engine generates an output cutset file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

3.16.5 FURST (Core & Fuel)

Description: Static & dynamic modeling

Validation: The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

Extent of Application: Mechanical design of core internals loads, deflections, and stress analysis for X300

3.16.6 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

3.16.7 MACCS v4

Description: The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (USNRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: MACCS will be used as part of the licensing basis events analysis in radiological consequences.

3.16.8 MCNPX v6

Description: Monte Carlo N-Particle Transport is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies and is developed by Los Alamos National Laboratory.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: MCNP will be used for performing criticality and shielding analyses. MCNP can be used in several transport modes: neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10-11 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 UO2 and (U,Gd)O2 fuel with (and without) additive material. PRIME03P is used for steady-state and transient licensing analysis as well as qualification cases of Recrystallized Annealed Zircaloy-2 cladding. Additionally, PRIME03P may be used with Stress-Relieved Annealed Zircaloy-4 cladding of either 70 % or 30 % cold work for qualification cases, but not for licensing analysis.

3.16.12 RAMP: GALE v3.2

Description: The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as “inputs”; GALE asks the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

3.16.13 RAMP: HABIT v2.2

Description: HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

3.16.14 RAMP: DandD v2.1

Description: A code for screening analyses for licence termination and decommissioning.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning" (Reference 3.12-15).

3.16.15 RAMP: GENII v2.10

Description: GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

3.16.16 RAMP: MILDOS v4

Description: Radiological dose commitment calculation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

3.16.17 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The USNRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.16.18 RAMP: PIMAL v4.1.0

Description: GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

3.16.19 RAMP: TurboFRMAC v2021 11.0.2

Description: Radiological Hazard evaluation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

3.16.20 RAMP: VARSKIN v1.0

Description: Occupational Dose Analysis Code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c), "Occupational does 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^2 of skin at a tissue depth of 0.007 centimeters (7 mg/cm^2).

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 235U, 239Pu, 238U and 241Pu; and decay heat from actinides 239U and 239Np. DECAY01A also includes the decay heat contribution from other Actinides (in addition to 239U and 239Np which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

3.16.27 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

APPENDIX 3F – COMPUTER PROGRAMS USED IN ENVIRONMENTAL AND RADIOLOGICAL ANALYSES SUPPORTING THE DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.17 Introduction

This appendix describes the major computer programs used in deterministic and probabilistic safety analyses. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.17.1 ADDAM Version 1.4.2

Description: The ADDAM (Atmospheric Dispersion and Dose Analysis Method) computer code computes the statistical distribution of radiation doses to an individual or population after the airborne release of radioactive material into the environment. See Chapter 15, Subsection 15.5.1.2.5 for a description.

Validation

Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application

See Chapter 15, Subsection 15.5. for extent of application.

3.17.2 DECAY v1

Description: DECAY01A calculates the decay heat power fraction after certain operation period and exposure of a fissile core.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: DECAY01A is an Engineering Computer Code developed by GE Hitachi Nuclear Energy (GEH) as a method to determine the decay heat (shutdown power) for BWR fuel. The code was created in response to USNRC IN96-39 (Reference 3.12-16) that brought attention to the extreme variation in decay heat calculations throughout the country. This was due to either overly conservative assumptions or a misapplication of the ANS Decay Heat Standards. The DECAY01A code has therefore gone to great lengths to assure both the validity and applicability of its calculations. DECAY01A works as a function of both the ANSI/ANS-5.1-1979 (Reference

3.12-17) or ANSI/ANS-5.1-1994 (Reference 3.12-17) decay heat standards used for domestic and advanced reactor designs respectively. These standards set forth values of decay heat from fission products of 235U, 239Pu, 238U and 241Pu; and decay heat from actinides 239U and 239Np. DECAY01A also includes the decay heat contribution from other Actinides (in addition to 239U and 239Np which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

3.17.3 RADTRAD (Analytical Methods/ Radiological Analysis)

Description: RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation.

Validation: The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

Extent of Application: The RADTRAD code is used for calculating accident doses, calculating transport of fission products inside the plant after an accident, performing filter loading calculations for post-accident.

3.17.4 RAMP: GALE v3.2

Description: The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as “inputs”; GALE asks the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

3.17.5 RAMP: HABIT v2.2

Description: HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

3.17.6 RAMP: DandD v2.1

Description: A code for screening analyses for licence termination and decommissioning.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512 (Reference 3.12-15).

3.17.7 RAMP: GENII v2.10 (Analytical Methods/Radiological Analysis)

Description: GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

3.17.8 RAMP: MILDOS v4

Description: Radiological dose commitment calculation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

3.17.9 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.17.10 RAMP: PIMAL v4.1.0

Description: GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

3.17.11 RAMP: TurboFRMAC v2021 11.0.2

Description: Radiological Hazard evaluation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

3.17.12 RAMP: VARSKIN v1.0

Description: Occupational Dose Analysis Code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c) {Reference 3.12-11), which states that the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm^2 of skin at a tissue depth of 0.007 centimeters (7 mg/cm^2).

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* VIe Control System in a wide range of applications, including control, safety integrity level, monitoring, and protection of assets. ToolboxST is one of the tools within ControlST, used for process configuration and diagnostics software for process, SIL, excitation and power conversion

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: For BWRX-300, the HFE team is using ToolboxST to provide early dynamic features and testing capability for the digital user interfaces designed using ActivePoint HMI/CIMPLICITY. The tool allows emulation of "live" screen features without the need for a plant simulation model driving the software. This allows early usability testing of digital user interfaces, as part of the HFE design testing and evaluation set of activities. The software is not used in production.

3.18.10 EPRI: SysImp v11

Description: Analysis of PRA Importance Measures. SysImp is a software tool used to calculate the importance of basic events, or collections of those events, in a risk model. SysImp is designed for risk models where components, equipment trains, and systems are represented by groups of basic events.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use SysImp to preform risk importance sensitivities, calculations, and grouping system importance. SysImp allows for deriving insights from risk importance rankings, estimating total plant risk given a specific change, and collective risk importance measures.

3.18.11 EPRI: UNCERT v11

Description: PRA Uncertainty Propagation analysis tool. Uncertainty Evaluation Tool (UNCERT). UNCERT can read the cut set or sequence data created from CAFTA and calculate the uncertainty of the cut set result.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use UNCERT to perform the parametric uncertainty calculations on the output cut sets. The UNCERT software will take a defined input (e.g., cut set file and associated CAFTA RR database) and perform the uncertainty analysis utilizing either a Monte Carlo or Latin Hypercube sampling method. The output will calculate the metrics for the cut set using that defined method.

3.18.12 GOTHIC v8

Description: GOTHIC is a procured software from Zachry Nuclear Engineering, Inc. for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: GOTHIC is used to perform a sensitivity analysis for the passive containment cooling system while developing the design.

3.18.13 MACCS v4

Description: The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (NRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: MACCS will be used as part of the licensing basis events analysis in radiological consequences.

3.18.14 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.18.15 SCALE v6

Description: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

3.18.16 TRACG v4

Description: TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

3.18.17 VTR.LMP

Description: Package of functions and data frames supporting VTR LMP applications. This package was developed using open-source code R. Currently only functions on a Mac platform.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project currently does not use this code package; however, developmental work is in progress to explore the application of this software to BWRX-

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.